CONTROL OF NUCLEAR PROPULSION PLANT POWER TRANSIENTS

Howard Maitland Chatham

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THESIS

CONTROL OF NUCLEAR PROPULSION PLANT
POWER TRANSIENTS

by

Howard Maitland Chatham

June 1975

Thesis Advisor: A. Gerba

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Control of Nuclear Propulsion Plant
Power Transients

by

Howard Maitland Chatham
Lieutenant, United States Navy
B.S., Iowa State University, 1968

Submitted in partial fulfillment of the

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TABLE OF SYMBOLS AND ABBREVIATIONS

SYMBOL	DEFINITION
c	Delayed neutron precursor concentration
c_{rp}	Reactor coolant specific heat
C	Normalized delayed neutron precursor concentration
C_{bp}	Reactor coolant thermal capacity
C_{bs}	Boiler water thermal capacity
C_{rf}	Fuel element thermal capacity
CRW	Control rod worth
ΔT	Temperature difference across the heat exchanger
DRW	Differential rod worth
h_b	Boiler overall heat transfer coefficient
h_r	Fuel element heat transfer coefficient
H_{fw}	Feed water enthalpy
H_s	Boiler steam enthalpy
K_{eff}	Effective neutron multiplication factor
l^*	Prompt neutron lifetime
m_p	Reactor coolant flow rate
m_s	Boiler steam flow rate
M	Normalized reactor coolant flow rate
n	Reactor neutron power
N	Normalized reactor neutron power
P	Normalized reactor thermal power
P_s	Boiler steam pressure
q	Reactor thermal power
t	Time

SYMBOL	DEFINITION
T_{ave}	Average coolant temperature in the heat exchanger
T_c	Cold leg temperature at the heat exchanger
T_{cav}	Average coolant temperature in the core
T_{cc}	Cold leg temperature at the reactor
T_{ch}	Hot leg temperature at the reactor
T_f	Average fuel temperature in the core
T_h	Hot leg temperature at the heat exchanger
T_s	Boiler steam temperature
α_f	Fuel temperature coefficient of reactivity
α_t	Coolant temperature coefficient of reactivity
β	Effective delayed neutron fraction
λ	Effective delayed neutron precursor decay constant
ρ	Reactivity
Ψ	Normalized steam flow

A dot $(\cdot)$ above a variable signifies the time derivative of that variable.

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I. INTRODUCTION

As the energy crisis becomes more pressing every day, nuclear power stands out as a most promising solution. Generation of electric power from nuclear energy was first accomplished in 1951 at the Atomic Enery Commission's (AEC) test facility in Idaho. Since then nuclear power stations have been built for commercial use in many countries and it has been estimated that three quarters of the total generating capacity of the United States will be from nuclear power by 2000 AD.

But power generation on land at present takes a second seat to marine plants. The United States alone has over one hundred ships with nuclear propulsion systems. Russia, England and France also have vessels with similar propulsion. The initial construction costs have limited most of these to naval craft, though West Germany and Japan have developed commercial ships with nuclear propulsion. The United States also built a commercial nuclear vessel, the Nuclear Ship (N.S.) Savannah, which has since been decommissioned.

The basic advantages of nuclear power versus fossil fuels are the small amounts of fuel required for substantial power generation, the relatively few pollutants generated, and the longevity of the fuel before refueling is necessary. Initial set up costs and costs of refueling are considerably more than for fossil fuel plants but the fewer number of plants for the same power generation and the longer lifetime of the plants more than counter-balance these costs.[8]

Marine nuclear power plants differ from stationery power generating

plants. Normally they are smaller physically and the output size required is less. In addition shipboard plants have more substantial power transients imposed on them. An electric generating station will have periods of peak load changes from normal steady state demands but these are insignificant compared to the propulsion orders that can arise in a maneuvering situation at sea. Power transients of up to eighty percent are not unlikely and a different mode of control than used on stationery plants is required.

Control systems on a nuclear plant keep system variables within proper limits and shut down the reactor when possible damage to the reactor or the environment is about to occur. Though the pressurized water reactor, the normal plant for shipboard use, is inherently stable it is also quick to respond. An automatic control system for this propulsion plant would minimize the possibility of human error, be able to cope with the quickness of response, and could also reduce the number of watchstanders required to operate the plant.

This thesis investigates the automatic control of a nuclear propulsion system to large power transients. To insure the validity of the system, the plant is modeled as closely as possible to that of the N.S. Savannah, the only unclassified power plant for which sufficient data was available. Similar load changes as applied in this study were conducted during power range testing on the N.S. Savannah with satisfactory results.[9]

The thesis is organized into sections as follows: first, the N.S. Savannah is described paying particular attention to the propulsion systems; next the reactor kinetics and thermodynamics equations

for the plant are developed. Following these are sections covering the results of the power transients to the plant without control systems and then with various control schemes applied. Finally the conclusions and recommendations for further study are discussed. An appendix follows presenting the comments and problems on the computer simulation.

II. N.S. SAVANNAH

A. THE SHIP

The Savannah was designed as a passenger-cargo ship and as such even with a conventional propulsion plant might not have been economical compared to bulk cargo ships. She was built to promote the acceptance of nuclear vessels throughout the world and to serve as a prototype for future merchant vessels. Because of the economical problems she was subsidized by the government but run by civilian concerns.

The ship began operations in early 1962. She made several trips overseas but political pressures in several countries against nuclear power curtailed many port visits. After much "showing of the flag" at home and abroad she was decommissioned in late 1971. Even though economically unsound at the time the ship was termed a success due to the performance of her propulsion plant, proving that commercial nuclear systems were practical.[1]

B. THE POWER PLANT

The basic nuclear power plant is divided into two system, the primary and the secondary.

1. The Primary System

This system generates the heat to boil water in the secondary system. It consists of the nuclear reactor, coolant pumps and piping, the heat exchanger and auxiliary systems. All of these are enclosed in a containment device which prevents the release of radioactive particles in case of some nuclear accident.

The reactor core is made up of uranium dioxide formed into thin fuel plates coated with a cladding material. Twenty-one control rods made of enriched boron-stainless steel are inserted between groups of fuel elements. The rods absorb neutrons and when fully inserted into the reactor sufficient neutrons are absorbed to prevent a self-sustaining nuclear reaction. As the rods are withdrawn less neutrons are absorbed thus allowing more fissions to take place.

The fission process generates thermal energy which is conducted through the cladding to water circulating through the reactor core. The water is kept pressurized to prevent boiling in the system. The water is forced from the reactor by coolant pumps in two loops. Midway in each loop is a counter-flow heat exchanger. This method of heat transfer to the secondary system is used to insure there is no possible contamination of the propulsion system by nuclear by-products in the coolant.

The major auxiliary systems consist of a pressurizer, coolant purification and make up systems. A steam buble is created by electric heaters in a pressure vessel piped to the coolant loop. The higher temperature in the pressurizer maintains the bubble in this system and not in the primary loop. The bubble allows for thermal expansion and contraction in the primary system while maintaining a relatively constant pressure. The purification system draws off twenty gallons perminute of coolant to run through an ion exchanger. The make up system stores the purified coolant and charges additional water as necessary to the primary loops to maintain volume.[8]

Data for analysis of this and the secondary system are included as Table I.[8,9]

TABLE I N.S. SAVANNAH PRIMARY AND SECONDARY CHARACTERISTICS

Reactor maximum operating power	70 Megawatts
Reactor flow (total)	8.64×10^{6} lb/hr
T_h (maximum operating power)	520°F
T_c (maximum operating power)	496°F
T_{ave}	508°F
T_f (maximum operating power)	1108°F
T_s (maximum operating power)	458°F
High power scram	130%
High T_h scram	540°F
Steam flow (total)	2.67×10^{5} lb/hr
Rod speed (maximum)	20 inches/minute
Fuel loading	
U^{235}	312.4 kg
U^{238}	6787.5 kg
Volume fractions	
water	0.567
control rods	0.041
fuel	0.247
stainless steel	0.145
α_f	-2.3×10^{-5}/°F
α_t	-1.88×10^{-4}/°F
β	75×10^{-4}
l^{*}	5×10^{-5} seconds
λ	0.1/seconds

2. <u>The Secondary System</u>

This system uses the heat of the primary system to create steam in a boiler connected to the heat exchanger. The propulsion system is very similar to most steam turbine systems. Steam from each boiler is piped to a propulsion turbine, a turbo-generator, the steam-driven feed pumps and auxiliary systems. The propulsion turbines also have a steam dump which allows steam to be by-passed around the turbine to the condenser. All condensed steam is pumped back to the boilers by feed pumps.

3. <u>The Plant Model</u>

Due to the complexity of a nuclear power plant some simplification of the model was undertaken in order to carry out this study. A single coolant loop primary plant was assumed. All primary auxiliary system effects were neglected. Complete mixing in the reactor and a constant differential rod worth were also assumed. The ship's electrical loads (hotel loads) and auxiliary steam loads were assumed to be constant regardless of the power level. Feed water temperature and flow rate were also assumed to be constant.

The actual variable available to the watchstanders on the Savannah were steam flow (m_s), coolant flow (m_p), boiler pressure and temperature (P_s,T_s), coolant hot leg, cold leg and average temperatures (T_h,T_c,T_{ave}), temperature differential across the heat exchanger (ΔT), neutron power (n), and control rod height. [8] In a saturated steam system P_s corresponds to T_s and this variable was not used in the simulation. As the reactor thermal power is equal to the temperature differential across the heat exchanger multiplied by the coolant specific heat and the coolant flow rate,

m_p and ΔT were combined into one term which when normalized became P. Normalization of steam flow and neutron and thermal power allowed comparison of the three power levels on the same plot. Other plots generated were control rod height and system temperatures. Control rod height was zero referenced at one hundred percent steady state power and represented the height of the group of rods which was controlling the reactor.

Figure 2-1 shows a simplified diagram of the power plant model.

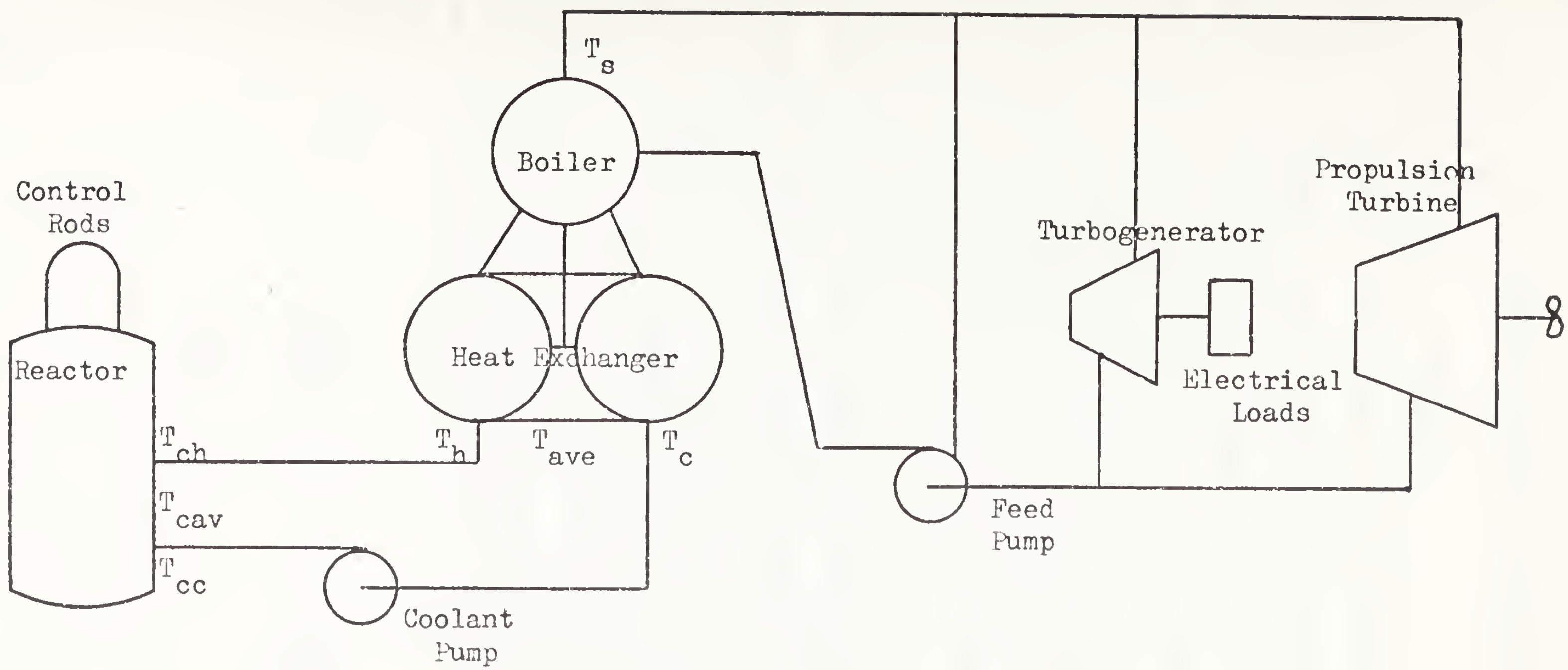

Figure 2-1 SIMPLIFIED DIAGRAM OF THE NUCLEAR POWER PLANT

III. REACTOR KINETICS AND THERMODYNAMICS

A. THE REACTOR KINETICS EQUATIONS

Reactor kinetics is the study of the time dependent behavior of a nuclear reactor. The first equation equates the rate of change of neutrons causing thermal fission to the rate of production of prompt, delayed and source neutrons causing thermal fission less the rate of loss of neutrons of all types. The other six equations relate the rate of change of the six delayed neutron precursors to the rate of production of each group of delayed neutron precursors less their rate of loss.

To simplify the reactor model the six groups of delayed neutron precursors are lumped into a single group.[5] The decay constant becomes dependent on the variation of K_{eff} from one, defined as reactivity, but has been assumed constant for this study with negligible effect on the results. In addition, since the contribution of source neutrons while operating above one percent power is insignificant, this term was also neglected. Thus the equations are as follows:

$$\dot{n} = \frac{n}{l^*}\cdot(1 - \beta) + \lambda\cdot c - \frac{n}{l^*\cdot K_{eff}} \qquad (3\text{-}1)$$

$$\dot{c} = \frac{n}{l^*}\cdot\beta - \lambda\cdot c \qquad (3\text{-}2)$$

For use in the computer simulation and power comparisons, normalization of these equations is convenient.[3] These normalized forms are shown on the next page with $N=n/n(0)$ and $C=c/c(0)$. The initial quantity is referenced to one hundred percent power.

$$\dot{N} = \frac{N}{l^* \cdot K_{eff}} \cdot (K_{eff} - 1) + \frac{\beta}{l^*} \cdot (C - N) \qquad (3\text{-}3)$$

$$\dot{C} = \lambda \cdot (N - C) \qquad (3\text{-}4)$$

K_{eff} is the ratio of the number of neutrons produced from fission in one generation to the preceeding generation. When K_{eff} equals one the reactor is said to be critical and neutron population is constant for successive generations. The power level is also constant and a steady state condition exists. Increasing K_{eff} causes increased neutron production, the supercritical condition. Decreasing K_{eff}, subcritical, causes a decrease in neutron production.[7]

Three major factors affect K_{eff}. These are average coolant temperature in the core, average fuel temperature and control rod height.[11] An increase in coolant temperature decreases the density of the water which causes less moderation, slowing down of fast neutrons, and fewer thermal fissions occur. Thus K_{eff} is reduced. Numerically this is computed by multiplying the change in temperature by a negative constant called the temperature coefficient of reactivity, in this case α_t. A similar effect takes place due to the temperature of the fuel and this negative reactivity coefficient is α_f.

The height of the control rods also changes reactivity. As rods are withdrawn more reactivity is "inserted" in the core; driving rods in reduces the reactivity. The amount of reactivity per inch of travel is called differential rod worth (DRW). For the Savannah the DRW was computed to be $7.0\text{x}10^{-4}$/inch.[8] The zero reference for rod height was set for one hundred percent steady state power and the control rod worth (CRW) is found by multiplying the DRW by the rod height.

Combining all these terms and noting that -0.120988 is the reactivity due to T_{ave} at 508°F and T_f at 1108°F, the following equation results:

$$K_{eff} = 1.120988 + \alpha_t \cdot T_{cav} + \alpha_f \cdot T_f + CRW \qquad (3\text{-}5)$$

B. THE THERMODYNAMIC EQUATIONS

1. Reactor Heat Transfer Equations

The total power generated in the reactor equals the thermal capacity of the fuel times the rate of change of the fuel temperature plus the heat transfer coefficient of the fuel times the temperature difference between the fuel and the coolant in the core.

$$q = C_{rf} \cdot \dot{T}_f + h_r \cdot (T_f - T_{cav}) \qquad (3\text{-}6)$$

And the latter term equals the thermal capacity of the coolant times the rate of change of the average coolant temperature in the core plus the coolant flow rate times the specific heat of the coolant times the temperature difference of the coolant across the reactor.

$$h_r \cdot (T_f - T_{cav}) = C_{bp} \cdot \dot{T}_{cav} + m_p \cdot c_{rp} \cdot (T_{ch} - T_{cc}) \qquad (3\text{-}7)$$

where $T_{cav} = (T_{ch} + T_{cc})/2.$ (3-8)

2. Heat Exchanger Equations

The coolant flow rate times the specific heat of the coolant times the temperature difference of the coolant across the heat exchanger equals the thermal capacity of the coolant times the rate of change of the average temperature of the coolant in the heat exchanger plus the heat transfer coefficient of the heat exchanger times the temperature difference between the coolant average temperature in the heat exchanger and the saturated liquid in the boiler.

$$m_p \cdot c_{rp} \cdot (T_h - T_c) = C_{bp} \cdot \dot{T}_{ave} + h_b \cdot (T_{ave} - T_s) \qquad (3\text{-}9)$$

And the latter term is equal to the thermal capacity of the boiler liquid times the rate of change of the saturation temperature in the boiler plus the steam flow rate times the difference of enthalpy between the steam and the feed water.

$$h_b \cdot (T_{ave} - T_s) = C_{bs} \cdot \dot{T}_s + m_s \cdot (H_s - H_{fw}) \qquad (3\text{-}10)$$

where $T_{ave} = (T_h + T_c)/2.$ (3-11)

3. Modification of the Point Equations

The above equations, representing the point model [10], do not account for the transport delays of six seconds in the coolant loop nor in the boiler. Letting

$$T_{ch}(t) = T_h(t+6) \qquad (3\text{-}12)$$

and

$$T_c(t) = T_{cc}(t+6) \qquad (3\text{-}13)$$

the time rate of change of T_h and T_{cc} can be evaluated by expansion by a Taylor series and neglecting second order and higher derivatives. In addition the enthalpy difference in equation (3-10) is approximately constant for this plant.

Evaluating constants as necessary, including coolant flow effects, and normalizing all power and flow terms [3], the following equations were used for the simulation:

$$\dot{T}_f = (600\ N - T_f + T_{cav})/25 \qquad (3\text{-}14)$$

$$\dot{T}_{cav} = (T_f - T_{cav} - 50\ M\ (T_{cav} - T_{cc}))/25 \qquad (3\text{-}15)$$

$$\dot{T}_h = M\ (2\ T_{cav} - T_h - T_{cc})/6 \qquad (3\text{-}16)$$

$$\dot{T}_{cc} = M\ (T_c - T_{cc})/6 \qquad (3\text{-}17)$$

$$\dot{T}_s = 70\ (P - \Psi)/3/M \qquad (3\text{-}18)$$

$$T_{ave} = (T_h + T_c)/2 \qquad (3\text{-}19)$$

$$T_c = ((2 - 0.48/M)\ T_h + 0.96\ T_s/M)/(2 + 0.48/M) \qquad (3\text{-}20)$$

$$P = 0.02\ (T_{ave} - T_s). \qquad (3\text{-}21)$$

IV. TRANSIENT ANALYSIS WITHOUT CONTROLS

A. TRANSIENT SELECTION

The transients selected for this study were an up power maneuver of twenty to eighty-five percent and a down power maneuver of one hundred to twenty percent. Both these transients were run on the N.S. Savannah and its control system handled both these transients, except as noted in section VI, satisfactorily.[8] The up power maneuver represents a load increse from normal "hotel loads" to a "full bell" (normal full speed) in ten seconds. The down power test represents the tripping of the propulsion throttle from maximum power, a three second transient.

In the up power case steady state conditions were assumed to have been reached when T_{ave} was within three degrees of its final value and neutron power within four percent power of its final value. For the down power case the temperature requirement remained the same but due to the more significant overshoots the power band was reduced to two percent.

It should be noted that due to the relatively low fuel enrichment of this reactor a significant α_f is generated. Without this term T_{ave} would have to return to its initial steady state value after any transient if no controls were applied. This characteristic is used in many power plants where more highly refined uranium can be used. In the Savannah this was not feasible but a constant T_{ave} control was still desired thus rod movement was used to counteract the fuel temperature effects.[13] Subsequent sections discuss the restrictions and advantages of the T_{ave} control system.

B. THE UP POWER TRANSIENT

The up power transient was run without any control system to determine the natural system response. Steady state was reached in ninety seconds after the start of the maneuver. As anticipated, as T_f increases with power, to maintain K_{eff} equal to one T_{ave} must decrease. ΔT increases as does the temperature difference between T_{ave} and T_s. Figures 4-1 and 4-2 present the plots of T_h, T_{ave}, T_c, and T_s and Ψ, N and P versus time.

Steady state values were as follows:

$T_h = 478.7°F$ $\Psi = 85.0\%$

$T_{ave} = 468.5°F$ $N = 81.3\%$

$T_c = 458.3°F$ $P = 84.6\%$

$T_s = 426.2°F$.

The peak overshoot of N was nine percent power. The minimum T_s was 423.0 F. It was noted that T_s was the first temperature to respond, followed in order by T_c, T_{ave} and T_h. This is due to the sudden change in steam demand draws energy from the boiler which reduces T_s. The water in the heat exchanger has more heat removed thus T_c decrease. T_{ave} necessarily decreases and since the thermal power of the reactor is less than the steam demand T_h increases relative to T_c. But due to the rapid decrease in T_c, T_h decreases also. Also noted was the fact that neutron power lagged the thermal power in the initial response by five seconds and it was over thirty seconds before it caught up with P.

C. THE DOWN POWER TRANSIENT

The natural response of this transient reached steady state in seventy seconds. Just the opposite of the up power transient occurred

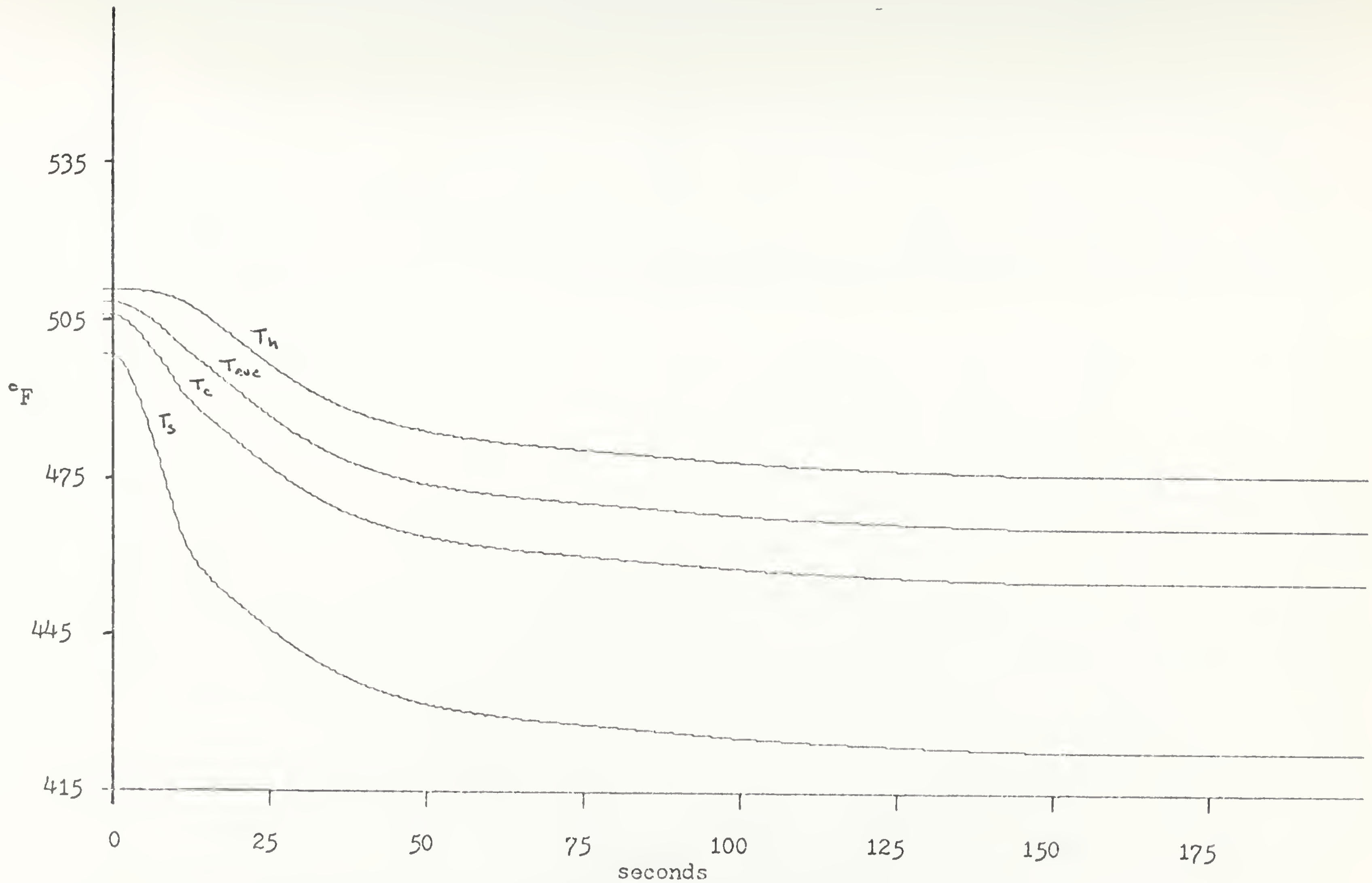

Figure 4-1 SYSTEM TEMPERATURES VERSUS TIME
UP POWER TRANSIENT WITHOUT CONTROLS

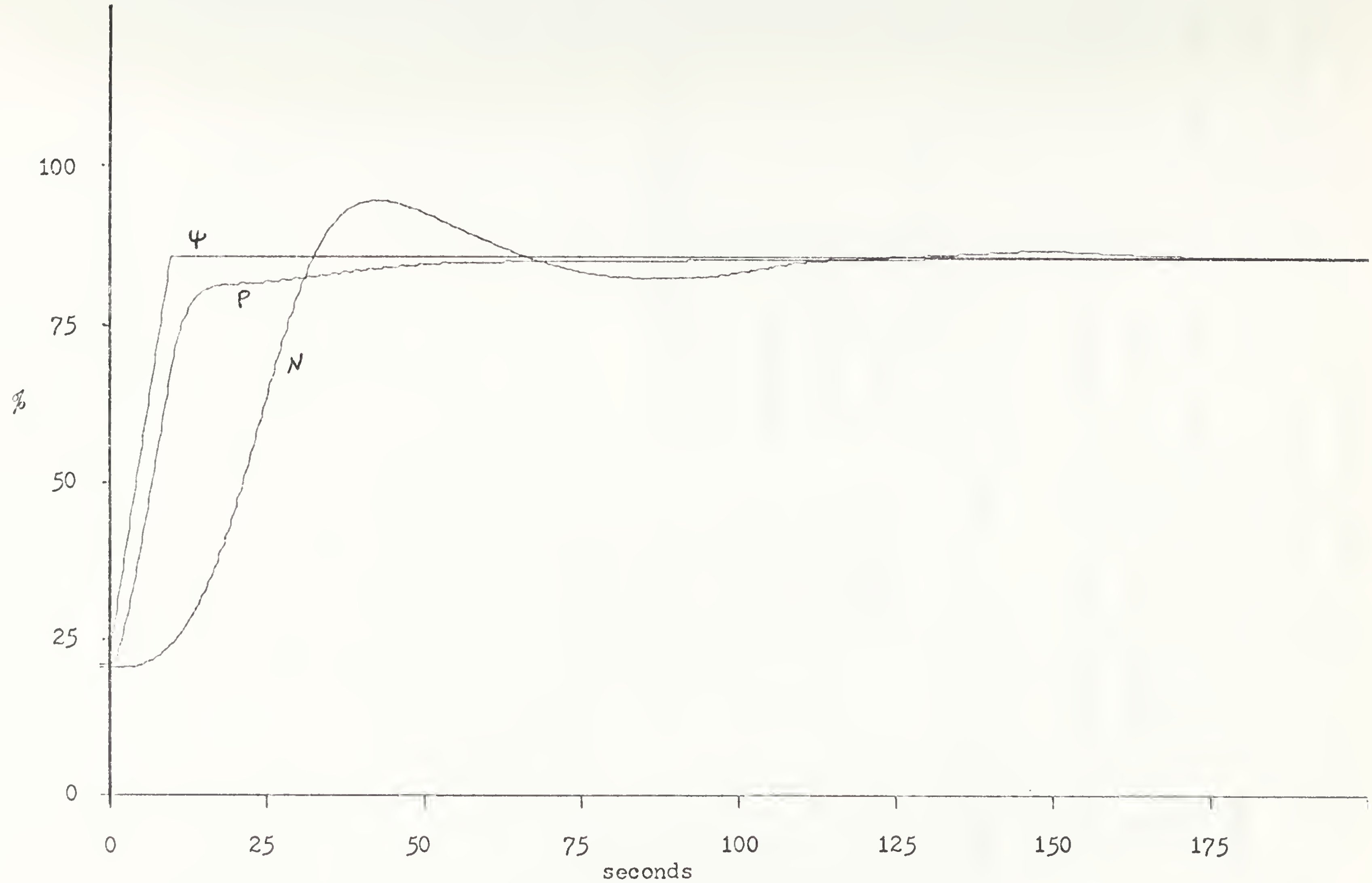

Figure 4-2 POWER LEVELS VERSUS TIME
UP POWER TRANSIENT WITHOUT CONTROLS

in the direction of the temperature values though the order of response stayed the same. As before the neutron power lagged thermal power. The initial delay was about three seconds and the crossover point was not reached until seventy seconds. The peak overshoot was only 2.3% power. (This represents a 11.5% overshoot of the steady state value as compared to only a 10.6% overshoot on the up power transient.)

Figures 4-3 and 4-4 show the results of this transient versus time. Steady state values were:

$T_h = 559.8°F$ $\quad$ $\psi = 20.0\%$

$T_{ave} = 557.3°F$ $\quad$ $N = 21.2\%$

$T_c = 554.8°F$ $\quad$ $P = 21.1\%$

$T_s = 546.8°F$.

The maximum T_h was 562.7°F.

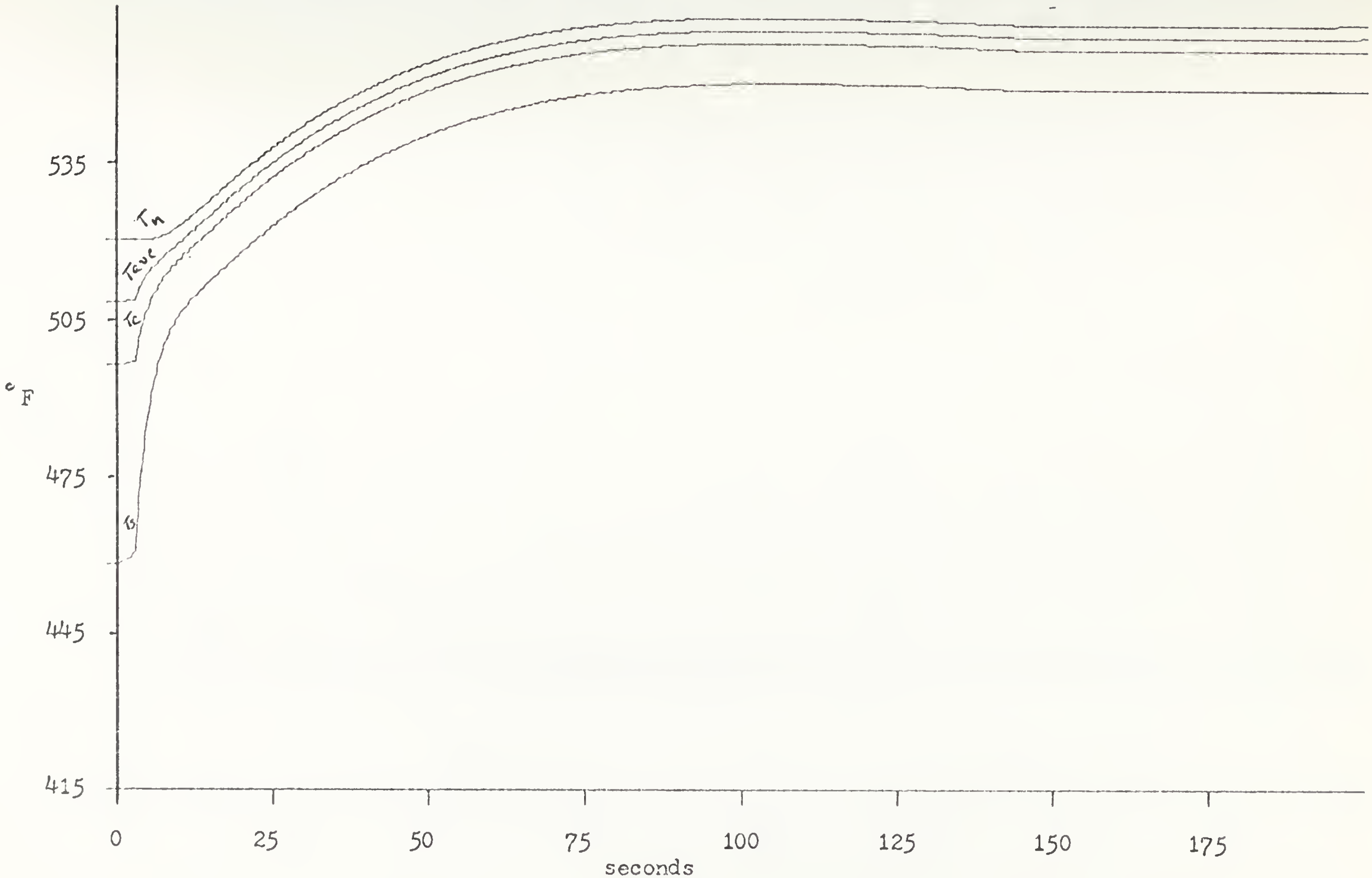

Figure 4-3 SYSTEM TEMPERATURES VERSUS TIME
DOWN POWER TRANSIENT WITHOUT CONTROLS

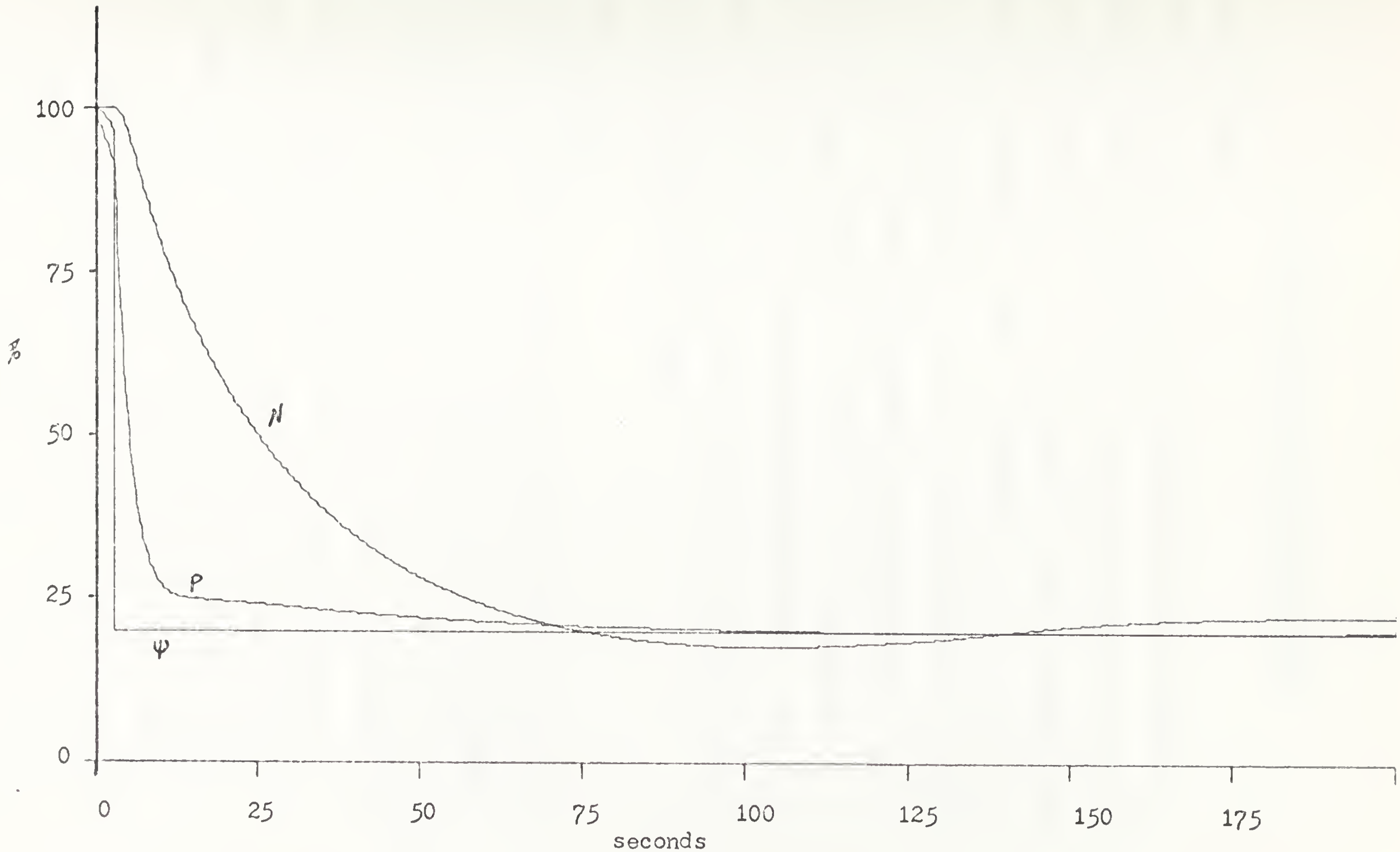

Figure 4-4 POWER LEVELS VERSUS TIME
DOWN POWER TRANSIENT WITHOUT CONTROLS

V. CONTROL FOR THE UP POWER TRANSIENT

A. BASIC CONTROL

The normal control system for a pressurized water reactor is to keep T_{ave} in a specified band. The primary reason for maintaining T_{ave} in this transient is to raise T_s. [12] Without controls it was noted that T_s dropped to 423°F. The higher T_s can be kept the more efficient the secondary plant will be. The Savannah used a control of $T_{ave} = 508 \mp 3$°F which was adopted in this study also. Since in this case T_{ave} decreased rod withdrawal was required to add reactivity to the core and raise system temperatures. The maximum rod withdrawal rate of twenty inches per minute was used to help speed up the response of the plant.

B. CONTROL LIMITS

In order to prevent excessive power generation in the core from causing a reactor accident, an automatic insertion of all control rods (scram) is initiated when N reaches 130%. Due to the tolerances involved in the power detection instruments and in the scram circuitry a ten percent power band below the scram set point was determined to be the maximum power allowable in this simulation.

To insure that this limitation would not be exceeded by any transient a test was run using a twenty to one hundred percent maneuver. When N reached 120% rod motion was terminated but power kept on increasing until it just reached 130%. It was evident that a lower power limit was required. Using a 110% limit the maximum power attained was 117%. Limits in between the two limits

either exceeded or were extremely close to 120% thus the 110% was adopted for further transients.

Attempting the test transient with this limit resulted in a maximum power of 118% but a further difficulty was encountered. Since rod motion can only counter less than 1.25°F per second, T_{ave} had originally dropped but was increasing and had reached 495°F when rod travel ceased. T_{ave} continued to increase and when power dropped below 110% rods were bumped to maintain that level. When T_{ave} reached its lower limit of 505°F rod motion again ceased but as T_{ave} rose above 511°F rod insertion commenced. By the time T_{ave} turned at 513°F power had been driven down to 59% and was still decreasing.

This over action necessitated a rod position limit. The rod worth corresponding to T_{ave} at 508°F and dependent on steam demand was used as the limit. Since T_f at steady state equals T_{ave} plus Ψ times 600, the limit was $0.0138 \cdot (\Psi - 1)$. This corresponds to a rod height of $19.7 \cdot (\Psi - 1)$ inches. Not allowing rod withdrawal past this point precluded the extreme power and temperature oscillations encountered above.

C. CONTROL SYSTEMS

1. T_{ave} Control

The 20 to 85% transient was run using the above limits and steady state was reached in 150 seconds. Steady state values were:

T_h = 515.3 F Ψ = 85.0%

T_{ave} = 505.0 F N = 87.9%

T_c = 494.7 F P = 85.3%

T_s = 462.3 F.

Figures 5-1, 2 and 3 show the plots of temperatures, power levels and control rod height versus time. The minimum T_s reached was 443.2°F, well above the minimum for the system response without controls. Maximum power was 118.2% and though N again lagged P initially, the crossover point was reached in only eighteen seconds, substantially less than in the previous run.

It was also noted that N was still in excess of Ψ and P after 200 seconds. This was due to the fact that the fuel with its slower time response than the coolant had not reached its final value. In order to raise T_f the neutron power must be greater than the thermal power and as the steady state value of T_f is approached the difference in power levels will diminish.

Initial rod motion was at the maximum rate of twenty inches per minute. But during the period of bumping rods to maintain the 110% N limit an average rate of 8.67 inches per minute was obtained.

2. Anticipatory Control

In an effort to reduce the response time, early rod withdrawal was initiated. As soon as steam demand exceeded neutron power by a five percent tolerance band the rod pull was commenced instead of waiting until T_{ave} exceeded its band.

In this case steady state was reached in 135 seconds even though rod travel was only initiated five seconds sooner than in the previous case. Steady state values were as follows:

T_h = 515.2°F Ψ = 85.0%

T_{ave} = 505.0°F N = 87.3%

T_c = 494.8°F P = 85.4%

T_s = 462.3°F.

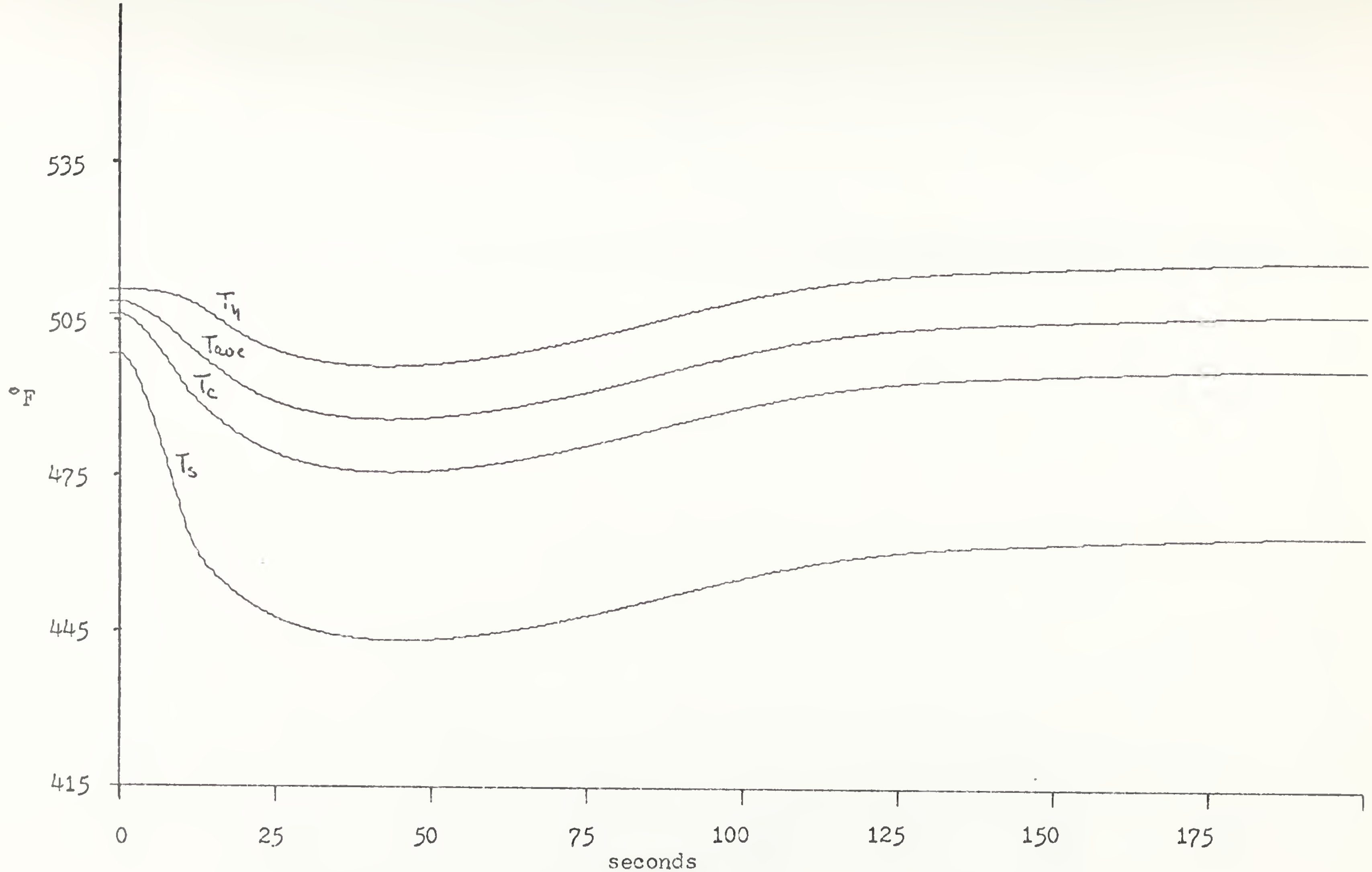

Figure 5-1 SYSTEM TEMPERATURES VERSUS TIME
UP POWER TRANSIENT, Tave CONTROL

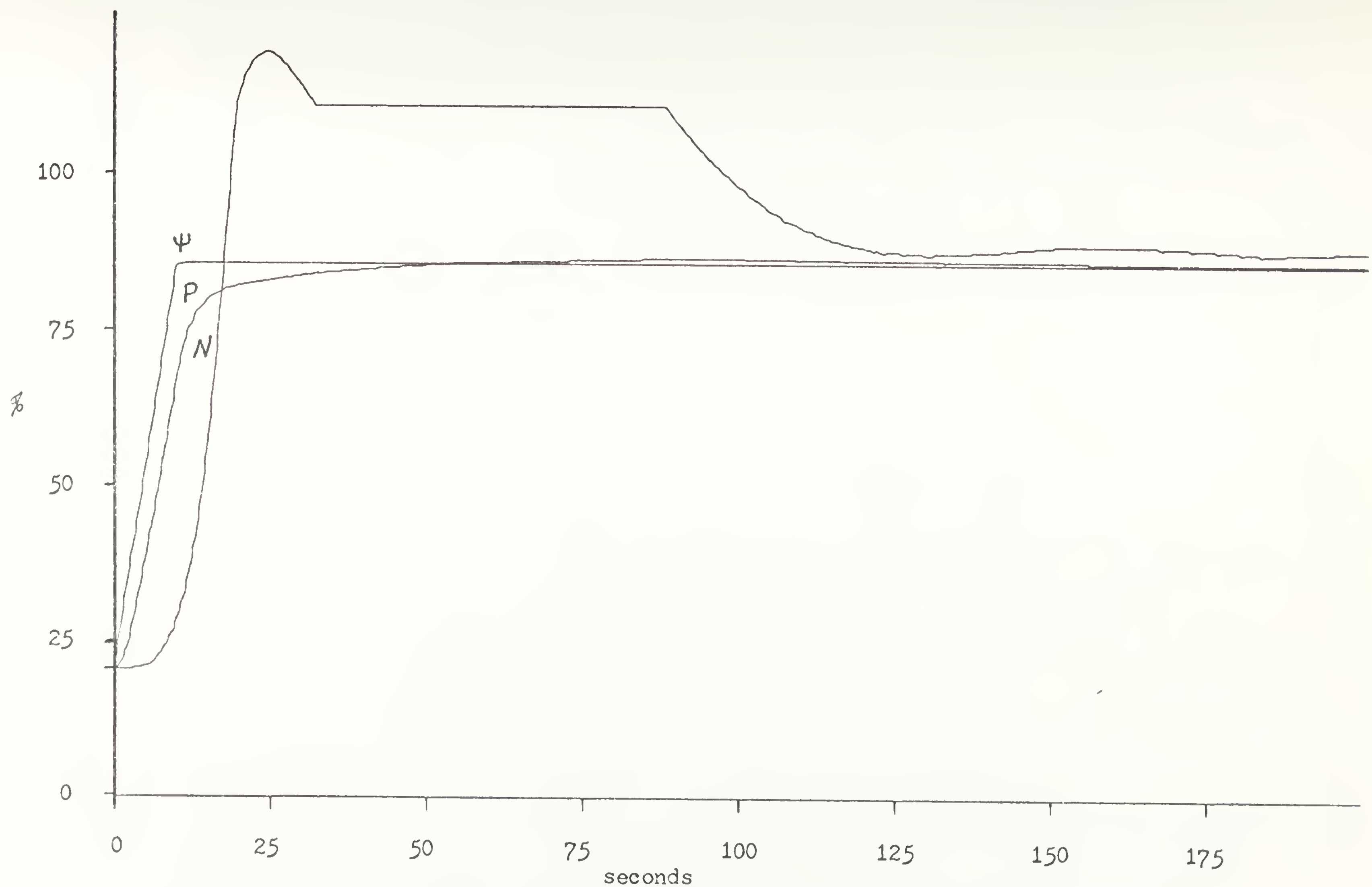

Figure 5-2 POWER LEVELS VERSUS TIME
UP POWER TRANSIENT, Tave CONTROL

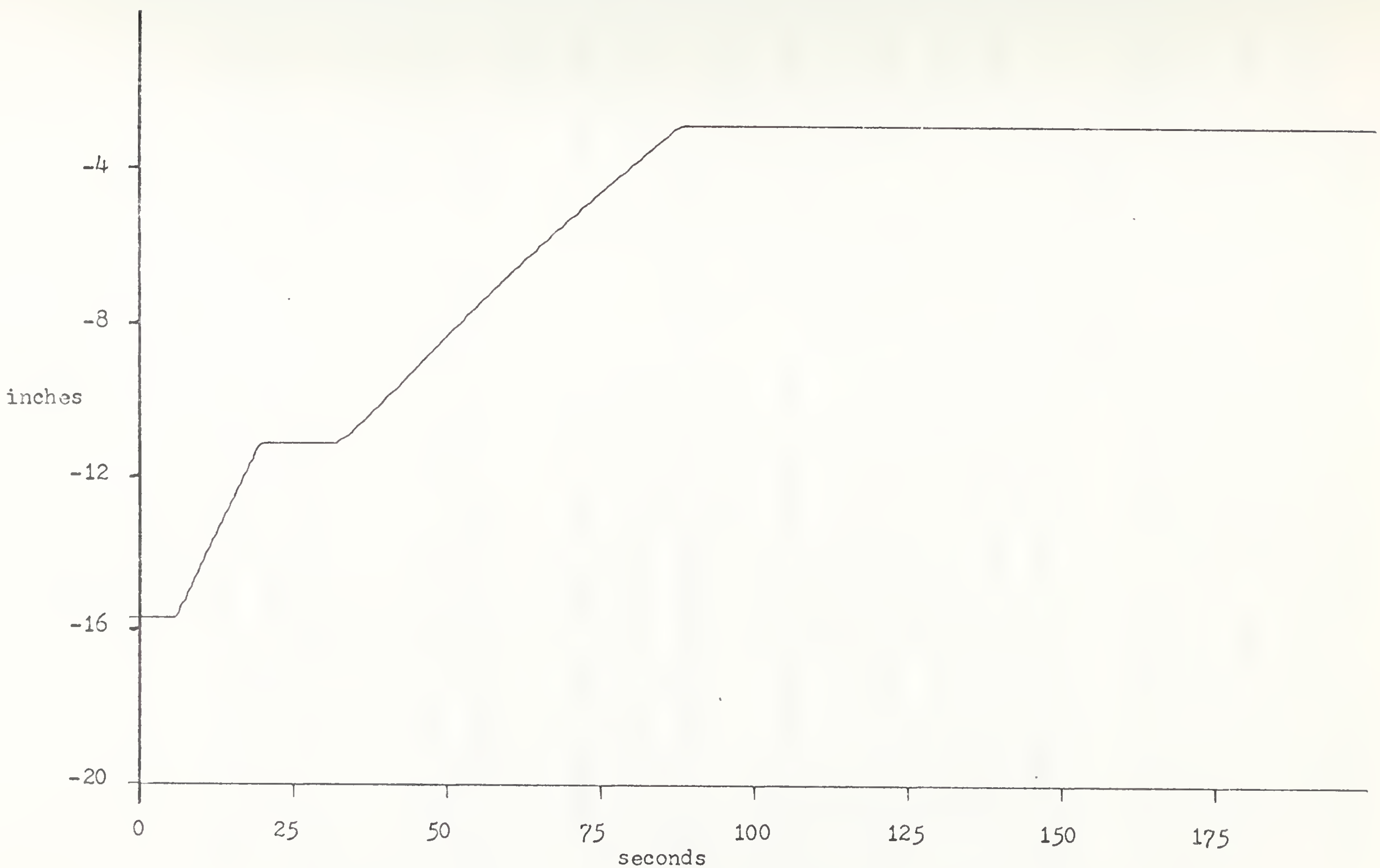

Figure 5-3 CONTROL ROD HEIGHT VERSUS TIME
UP POWER TRANSIENT, Tave CONTROL

Minimum T_s was only 446°F in this case, an increase above the previous run. Maximum power reached 119% and the crossover point for N and P was only fifteen seconds. Again the initial rod withdrawal was at twenty inches per minute whereas the subsequent average rate was 9.12 inches per minute. See figures 5-4,5 and 6.

3. Discrete Flow Change

Many reactor plants have the capability of two speed coolant pumps. At high power levels, normally above 50%, the pumps are shifted to the faster speed usually twice the flow rate of the slow speed. This reduces the ΔT and with T_{ave} constant decreases T_h. As this is a desirious result as will be discussed for the down power transient, an investigation of this flow change was conducted for the up power maneuver too.

A simulation was run without temperature control but with the coolant pump shift at 50% N and steady state was reached in ninety-four seconds, only slightly slower than without the pump shift.

Another run using the T_{ave} control and the pump shift reached steady state in 156 seconds only six seconds slower than without the flow change.

Combining all three control schemes steady state was reached in 141 seconds. This also was only six seconds slower than the anticipatory control case. Steady state values in this case were:

T_h = 510.1°F Ψ = 85.0%

T_{ave} = 505.0°F N = 87.9%

T_c = 499.9°F P = 85.8%

T_s = 462.1°F.

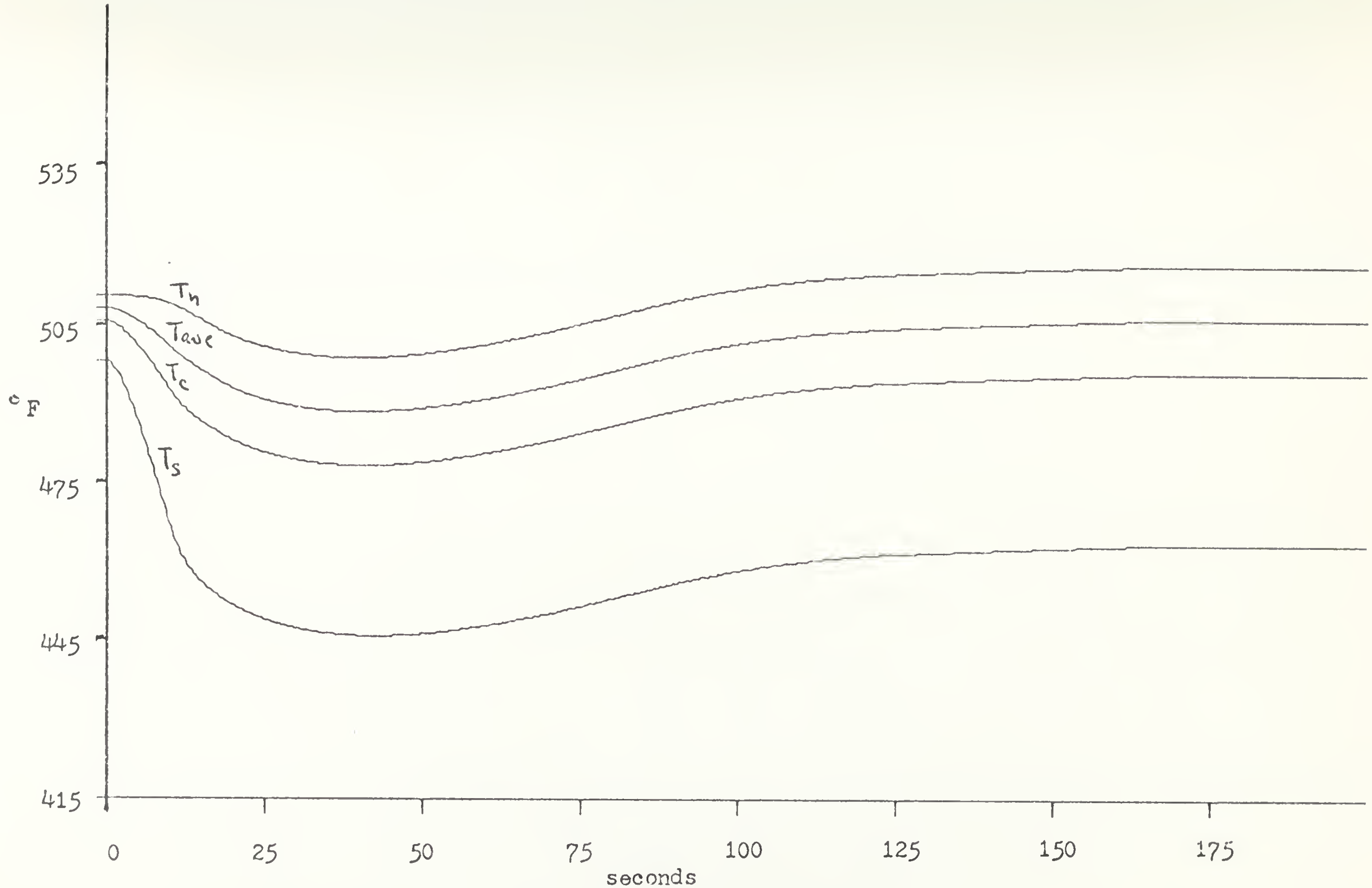

Figure 5-4 SYSTEM TEMPERATURES VERSUS TIME
UP POWER TRANSIENT, ANTICIPATORY CONTROL

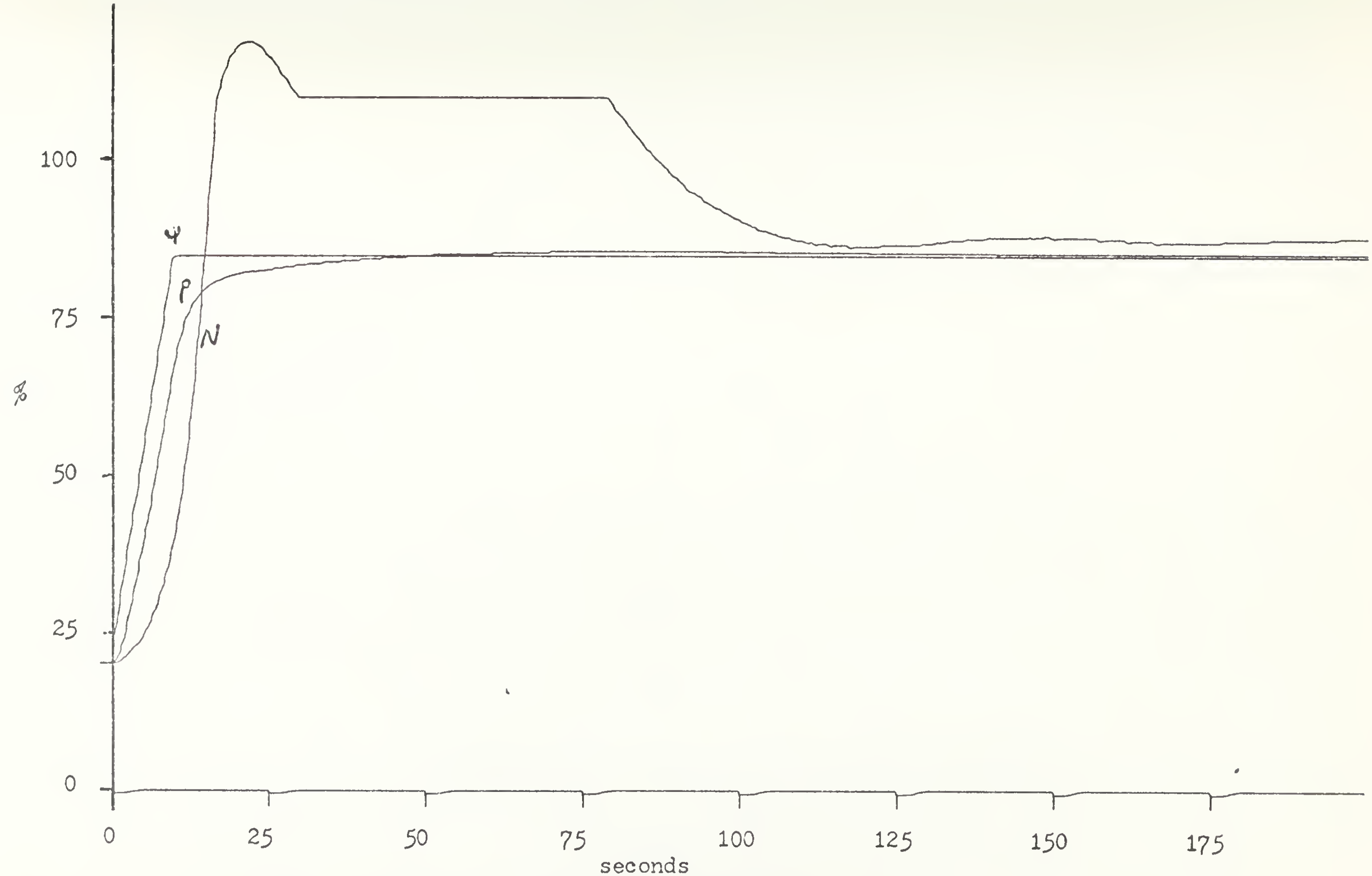

Figure 5-5 POWER LEVELS VERSUS TIME
UP POWER TRANSIENT, ANTICIPATORY CONTROL

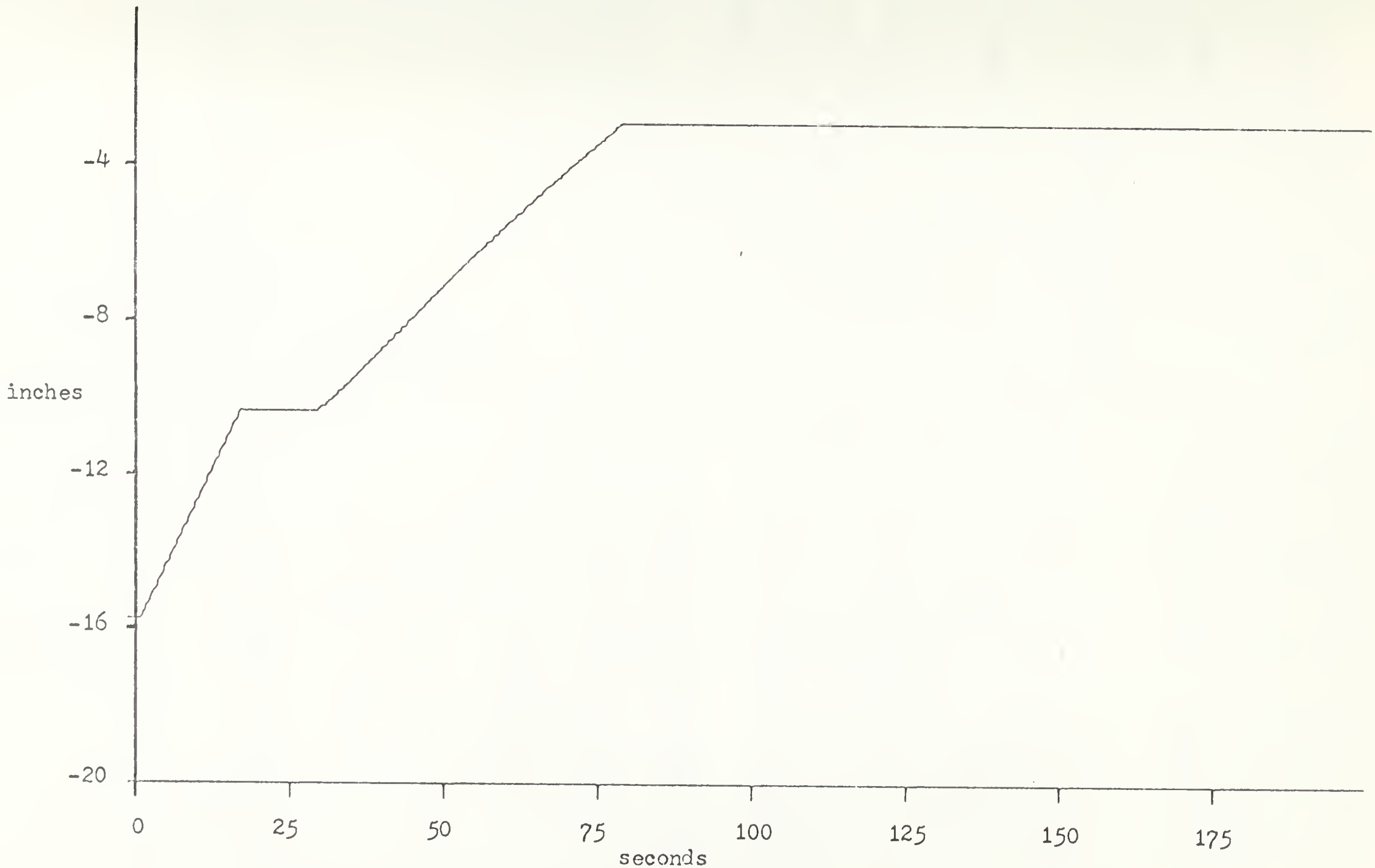

Figure 5-6 CONTROL ROD HEIGHT VERSUS TIME
UP POWER TRANSIENT, ANTICIPATORY CONTROL

The system showed an increased T_s minimum of 447.1°F and a reduced power level maximum of 117%. More of an overshoot in P was observed than before, 2.5% power, and the second rod withdrawal average rate was reduced to 8.66 inches per minute. Figures 5-7, 8 and 9 show the pertinent data.

4. <u>Continuous Flow Change</u>

In an attempt to reduce the response time a continuous flow change was assumed. The coolant pump was assumed to be variable speed, normal flow rate at 0% power and twice that rate at 100% steam flow. In between a linear relationship was assumed thus M equaled Ψ plus one. Steam flow was used in this control because N was known to exceed 100% whereas steam flow was limited by the boiler and turbine construction.

In this case the transient system reached steady state in 138 seconds, three seconds faster than in the previous case. Minimum T_s was 447.8°F and maximum power was 117%. Figures 5-10, 11 and 12 show the variable plots and steady state values were:

T_h = 510.5°F $\quad\quad$ Ψ = 85.0%

T_{ave} = 505.0°F $\quad\quad$ N = 87.8%

T_c = 499.5°F $\quad\quad$ P = 85.7%

T_s = 462.1°F.

The P, N crossover point was just over fifteen seconds and the second rod withdrawal average rate was 8.76 inches per minute.

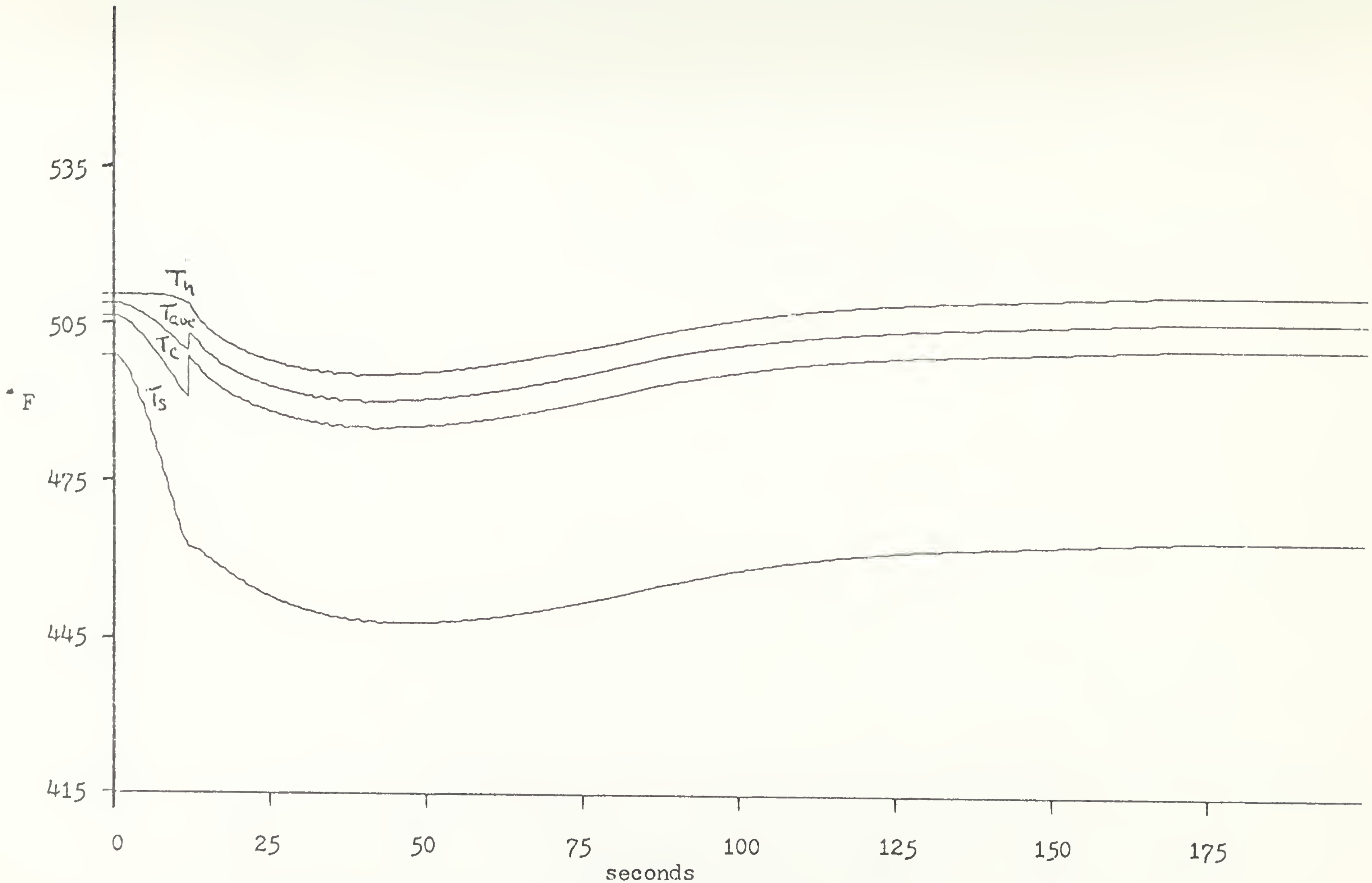

Figure 5-7 SYSTEM TEMPERATURES VERSUS TIME
UP POWER TRANSIENT, DISCRETE FLOW CHANGE

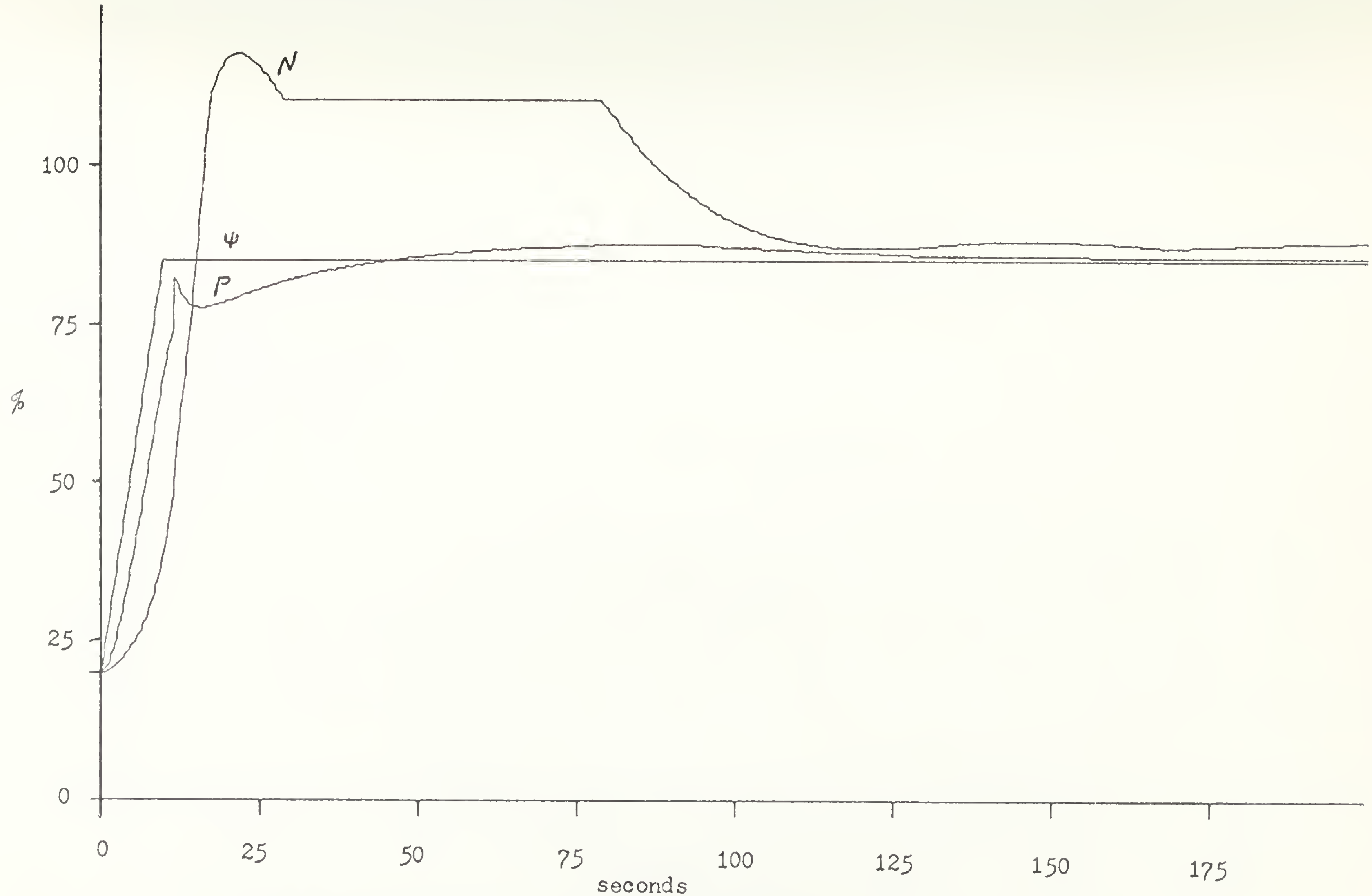

Figure 5-8 POWER LEVELS VERSUS TIME
UP POWER TRANSIENT, DISCRETE FLOW CHANGE

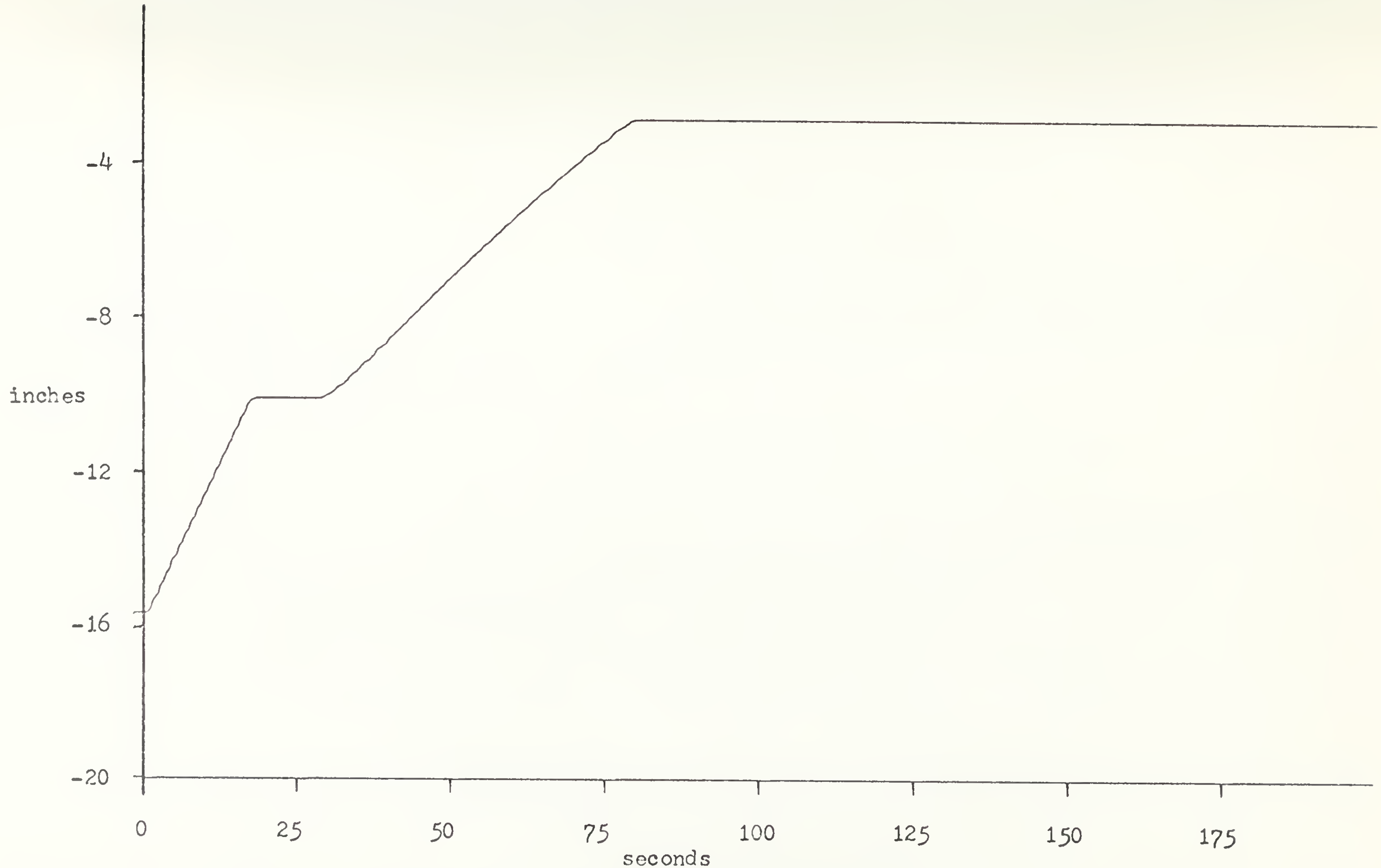

Figure 5-9 CONTROL ROD HEIGHT VERSUS TIME
UP POWER TRANSIENT, DISCRETE FLOW CHANGE

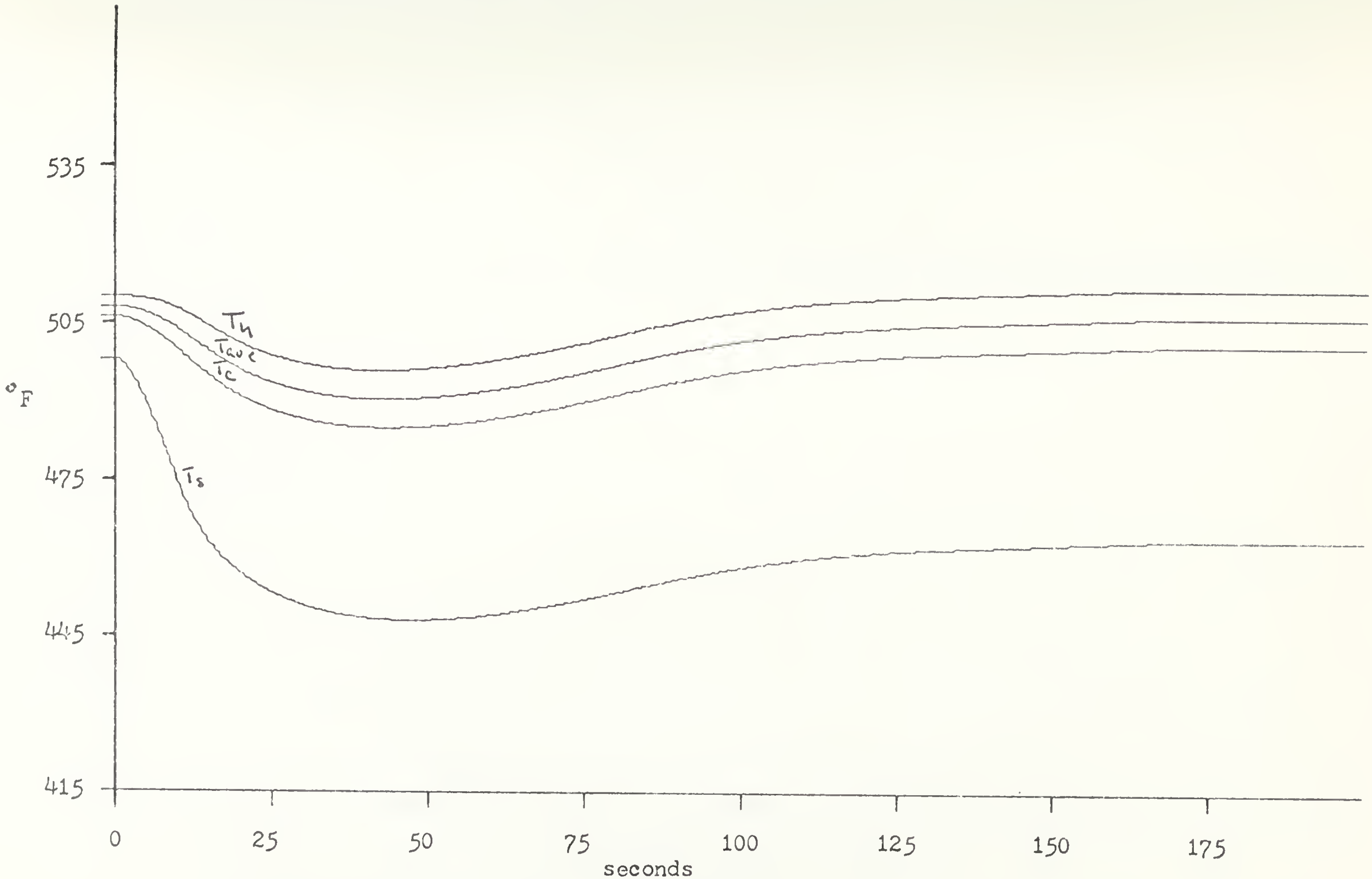

Figure 5-10 SYSTEM TEMPERATURES VERSUS TIME
UP POWER TRANSIENT, CONTINUOUS FLOW CHANGE

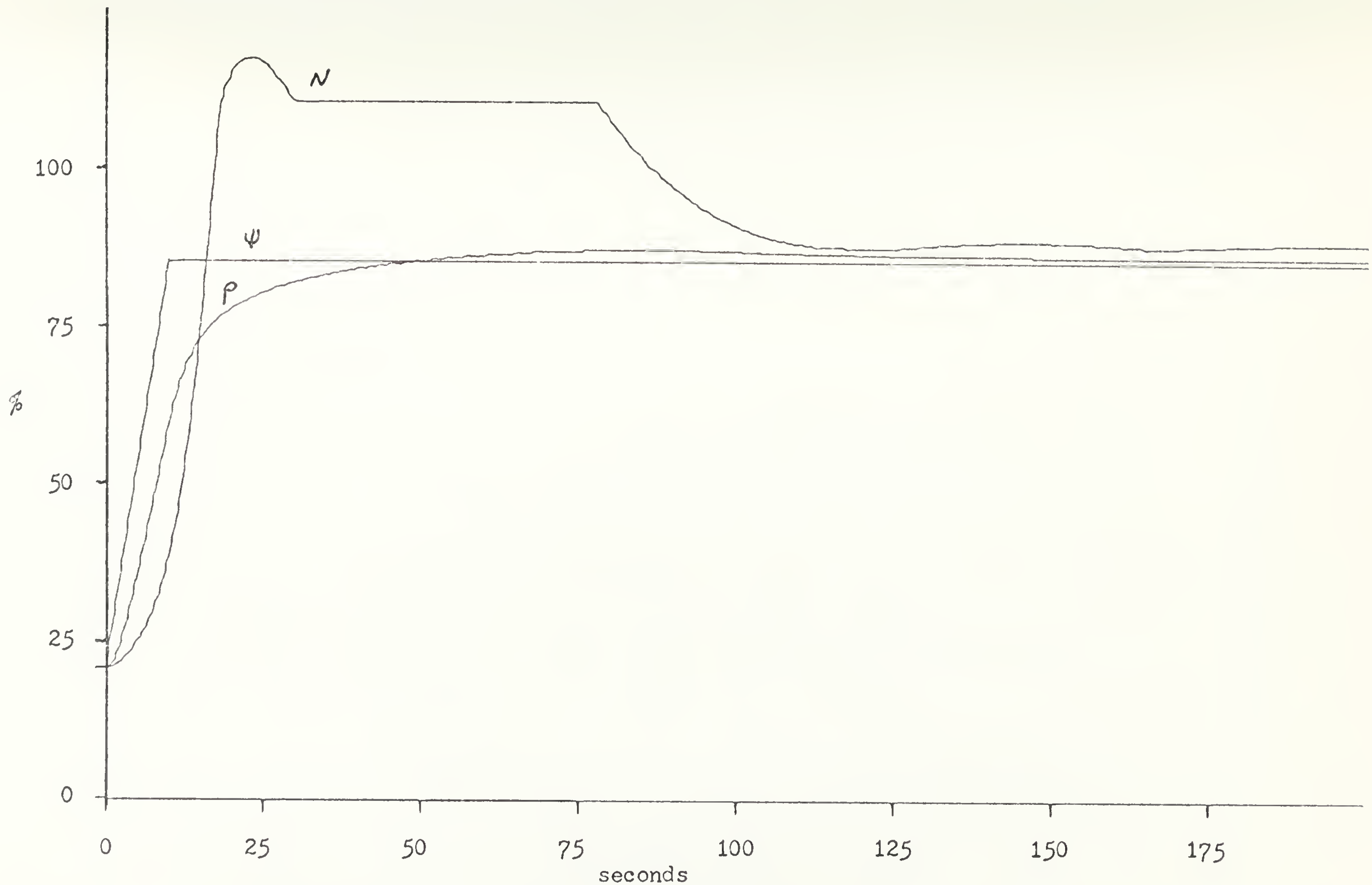

Figure 5-11 POWER LEVELS VERSUS TIME
UP POWER TRANSIENT, CONTINUOUS FLOW CHANGE

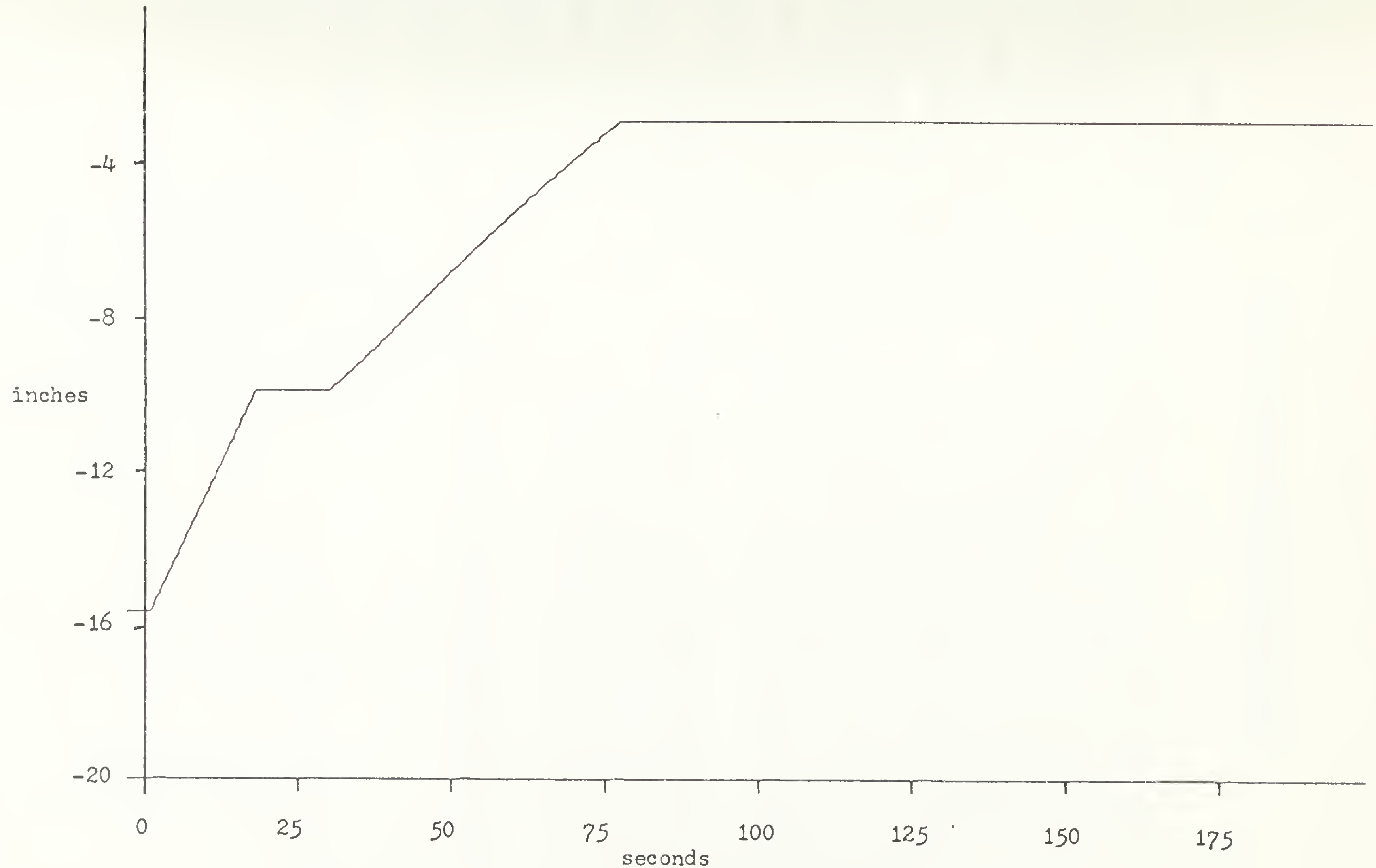

Figure 5-12 CONTROL ROD HEIGHT VERSUS TIME
UP POWER TRANSIENT, CONTINUOUS FLOW CHANGE

VI. CONTROL FOR THE DOWN POWER TRANSIENT

A. BASIC CONTROL

As with the up power transient the control for the down power transient is based on maintaining $T_{ave} = 508 \mp 3°F$. But instead of attempting to raise T_s, decreasing T_h is one of the primary concerns. High T_h could cause fuel cladding burnout and release of fission products to the coolant. The N.S. Savannah had a high T_h scram at 540°F and this limit was adopted for this study. As temperature indicators are more accurate than power instruments the large tolerance band needed for the high power scram is not needed for the high T_h scram.

Another concern is power overshoot, i.e., power decreasing below the twenty percent steam demand. Many reactor accidents are more severe at low power levels and levels of less than one percent virtually shut down the reactor.

Since T_{ave} needed to be decreased, as seen in figure 4-3, rods were inserted into the core.

B. CONTROL PROBLEMS

T_{ave} control was attempted using several rod speeds and control limits. Using the maximum speed power decrease to less than one percent. A minimum power limit of 15% was established to allow rod motion after N had past Ψ thus continuing to reduce T_{ave} but keeping power above several percent. To preclude the temperature and power oscillations noted in the up power transients the same rod limitation was imposed. Thus rods could not be inserted past

19.7 (ψ - 1) inches, the steady state height for the rods when T_{ave} was 508°F.

Even using these limits either the reactor reached power levels less than one percent or T_h exceeded 540°F. This was also the case on board the Savannah. To counteract this situation an automatic steam dump was designed. As steam pressure suddenly increased, as in a down power transient, the steam dump was automatically opened fully into the condenser. This throttle then could be shut at a slower rate while allowing the propulsion turbine to maintain its proper load. [8]

No mention of a specified closure rate of the steam dump throttle could be found in the literature. Various length transients were simulated at the slowest rod speed, five inches per minute. A sixty second transient was selected as being easily handled mechanically though a maximum T_h of 550°F was obtained.

Using this maneuver several rod speeds were tried to decrease T_h without shutting down the reactor by too rapid insertion of reactivity. Ten inches per minutes was found to reduce T_h just below 540°F while keeping power over seven percent. This rate was used in all down power controls.

C. THE NEW TRANSIENT

The sixty second down power transient was run without control systems to establish reference data. Steady state was reached in 108 seconds. The variable plots are shown in figures 6-1 and 6-2. As noted in the up power transients neutron power lagged thermal power initially. The crossover point was reached in 115 seconds with an insignificant neutron power overshoot. The steady

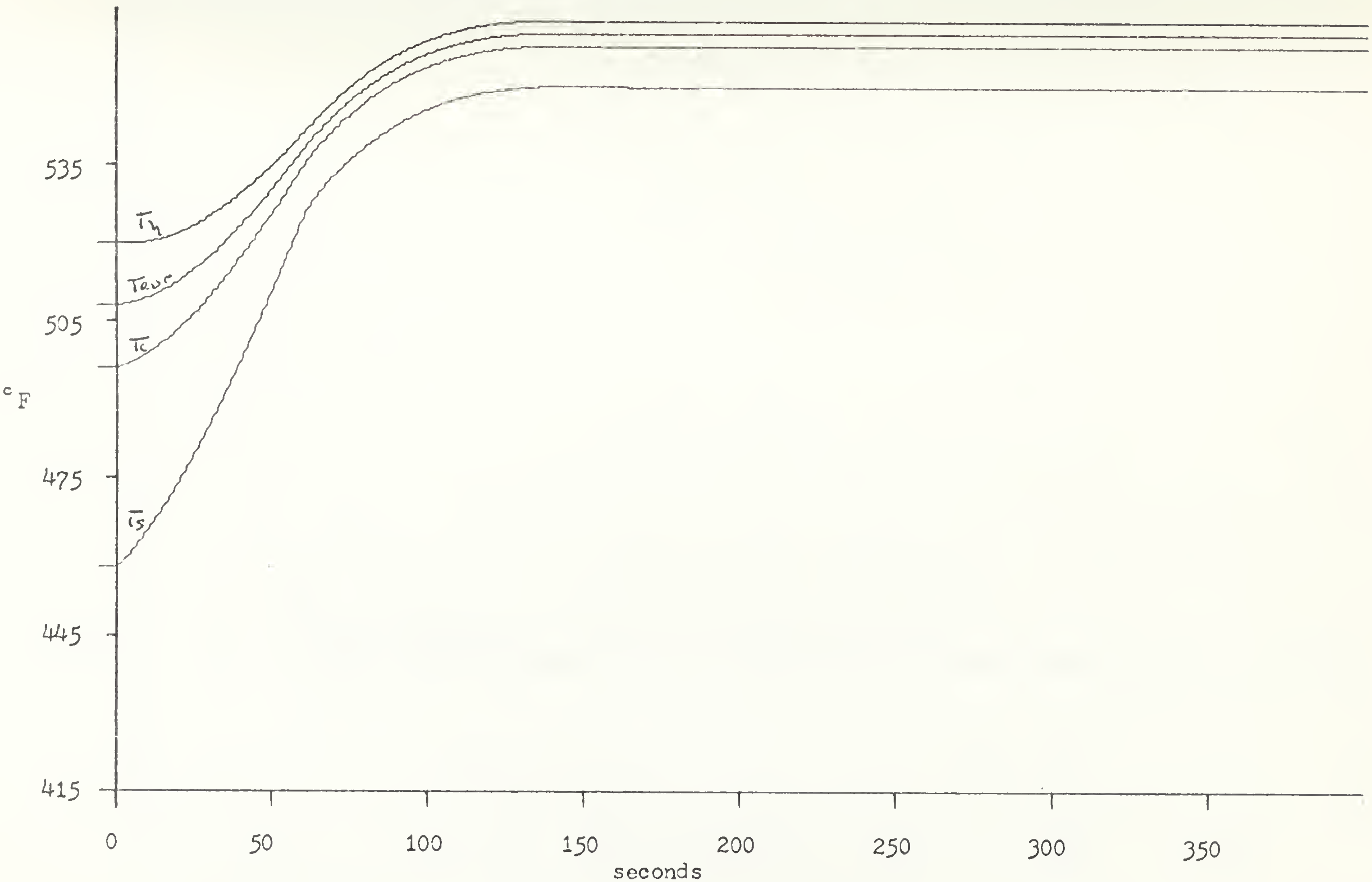

Figure 6-1 SYSTEM TEMPERATURES VERSUS TIME
MODIFIED DOWN POWER TRANSIENT WITHOUT CONTROLS

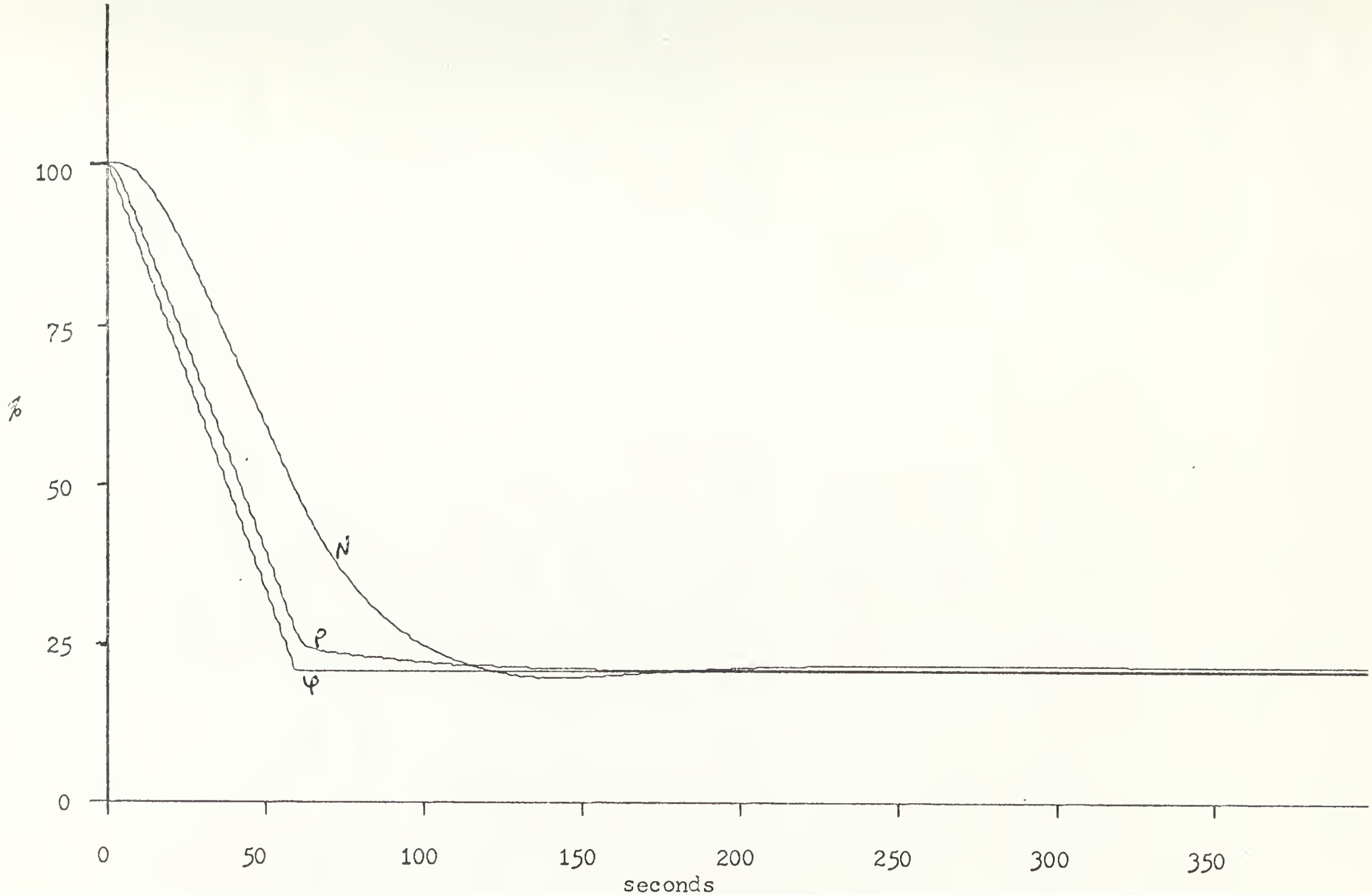

Figure 6-2 POWER LEVELS VERSUS TIME
MODIFIED DOWN POWER TRANSIENT WITHOUT CONTROLS

state values of temperature remained the same as for the three second transient while N was 22.0% and P was 21.0%.

D. CONTROL SYSTEMS

1. T_{ave} Control

With this control system the steady state conditions were reached after 356 seconds. Figures 6-3, 4 and 5 show the pertinent plots. Steady state values were:

$$T_h = 510.4^\circ F \qquad \Psi = 20.0\%$$
$$T_{ave} = 508.0^\circ F \qquad N = 18.0\%$$
$$T_c = 505.6^\circ F \qquad P = 19.8\%$$
$$T_s = 498.1^\circ F.$$

Maximum T_h was just under 540°F and minimum N was 7.4%. N reached P after about thirty seconds and maintained about the same rate of decrease until thermal power reached 22%. Because of the longer and less peaked power overshoot the subsequent rod motion was not linear at first. After twenty seconds an average rate of 1.63 inches per minute was obtained while the initial rod travel had been at a rate of ten inches per minute.

It should also be noted that the neutron power exhibits oscillations of up to two percent power even after steady state. Though T_h, T_{ave}, T_c, and T_s have stabilized T_f is still decreasing and as N is affected the most by T_f and is more responsive to reactivity changes at low powers these oscillations will continue for several minutes.

2. Anticipatory Control

In this case rod insertion was commenced when steam demand was less than neutron power by five percent. The response time

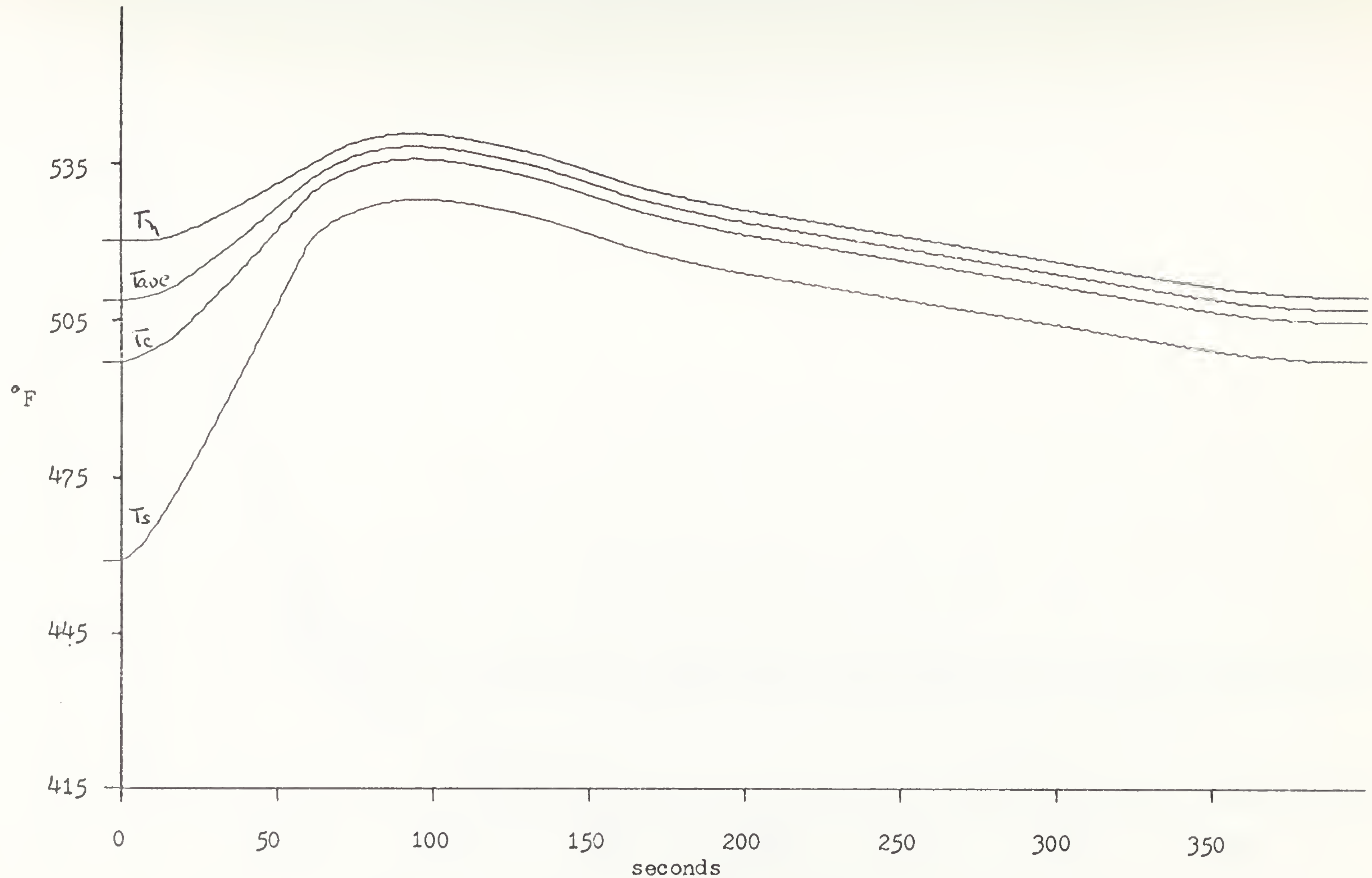

Figure 6-3 SYSTEM TEMPERATURES VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, Tave CONTROL

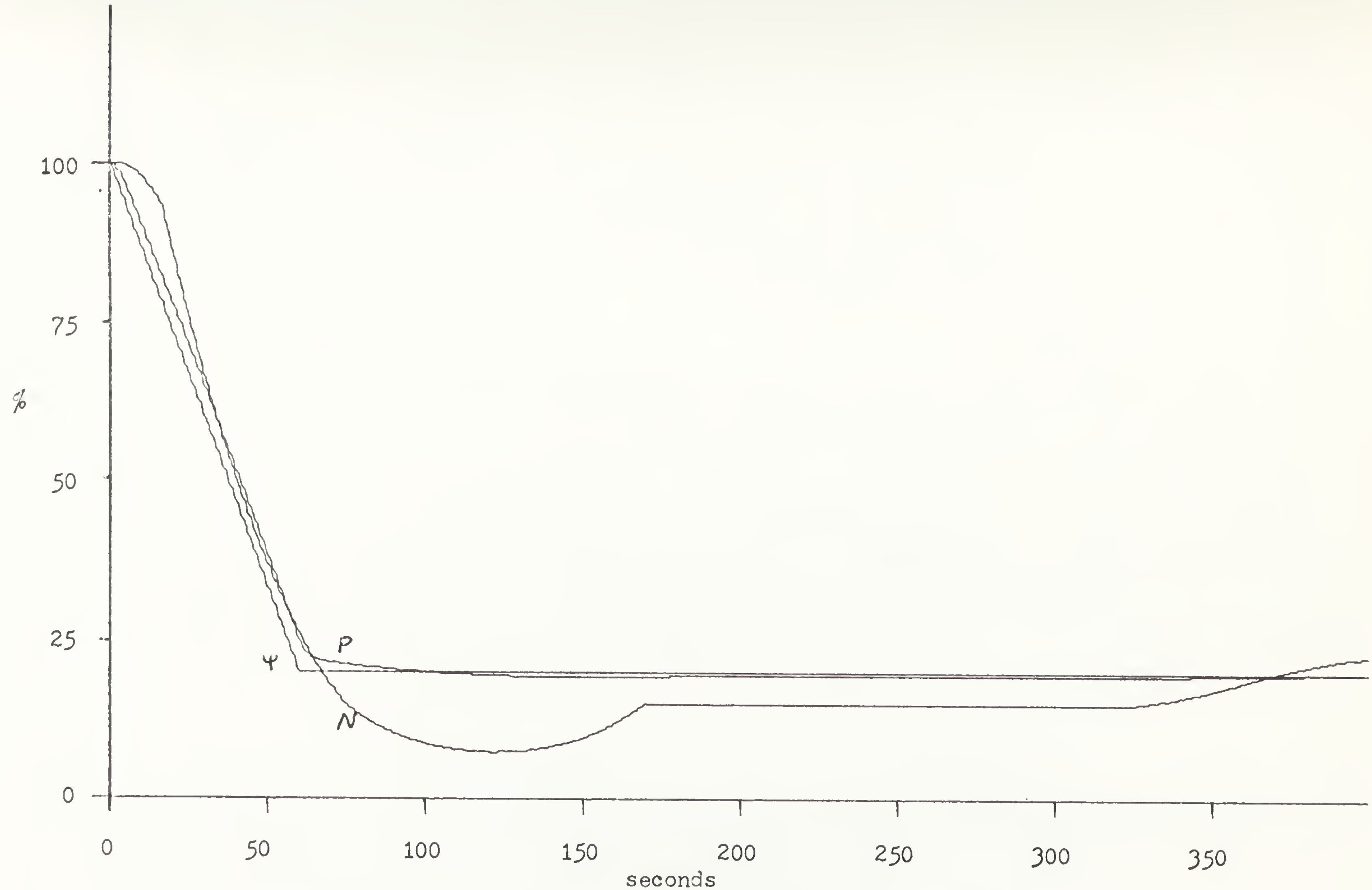

Figure 6-4 POWER LEVELS VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, Tave CONTROL

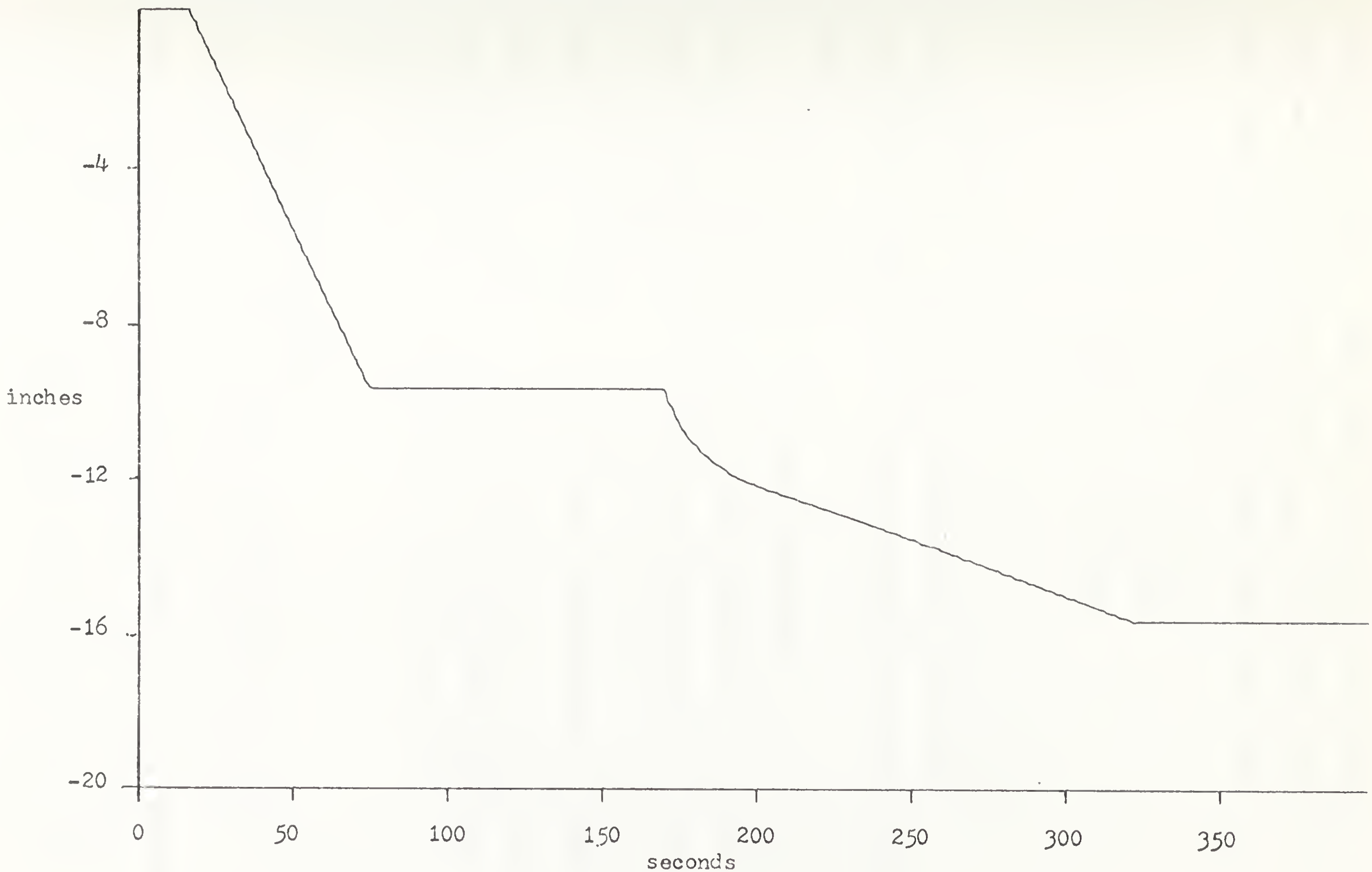

Figure 6-5 CONTROL ROD HEIGHT VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, Tave CONTROL

was greatly reduced. Steady state was reached in 315 seconds, over forty seconds faster than with the T_{ave} control only. Figures 6-6, 7 and 8 show the relevant plots. Steady state values were:

$$T_h = 510.5°F \qquad \Psi = 20.0\%$$
$$T_{ave} = 508.1°F \qquad N = 18.0\%$$
$$T_c = 505.7°F \qquad P = 19.8\%$$
$$T_s = 498.2°F.$$

T_h maximum was reduced to 536.5°F and minimum power was increased to 7.7%. Rod insertion began with in four seconds. N decreased at a more rapid rate than P for most of the steam demand change.

The rate change in the initial rad response is due to rod initiation at 95% steam demand but since N followed Ψ so closely, just five percent or less difference until steam demand stabilized, only a 6.05 inches per minute average rate was abtained until T_{ave} reached 511 F. Even so, $9.31 \times 10^{-4}\ \rho$ was inserted before rod motion would have been started in the T_{ave} control mode. As with the T_{ave} control the subsequent rod motion was not linear until well after N returned to 15% and then maintained a rate of 1.77 inches per minute.

3. <u>Discrete Flow Change</u>

By using a pump shift when increasing power the reactor T is reduced. Thus T_h is reduced when operating at high power levels. Down shifting pumps when power decrease to 50% was attempted and the response time was further reduced to 309 seconds. Figures 6-9, 10 and 11 show the power, temperature and control rod height plots. The maximum T_h was 535.8°F and power reached 7.7% again.

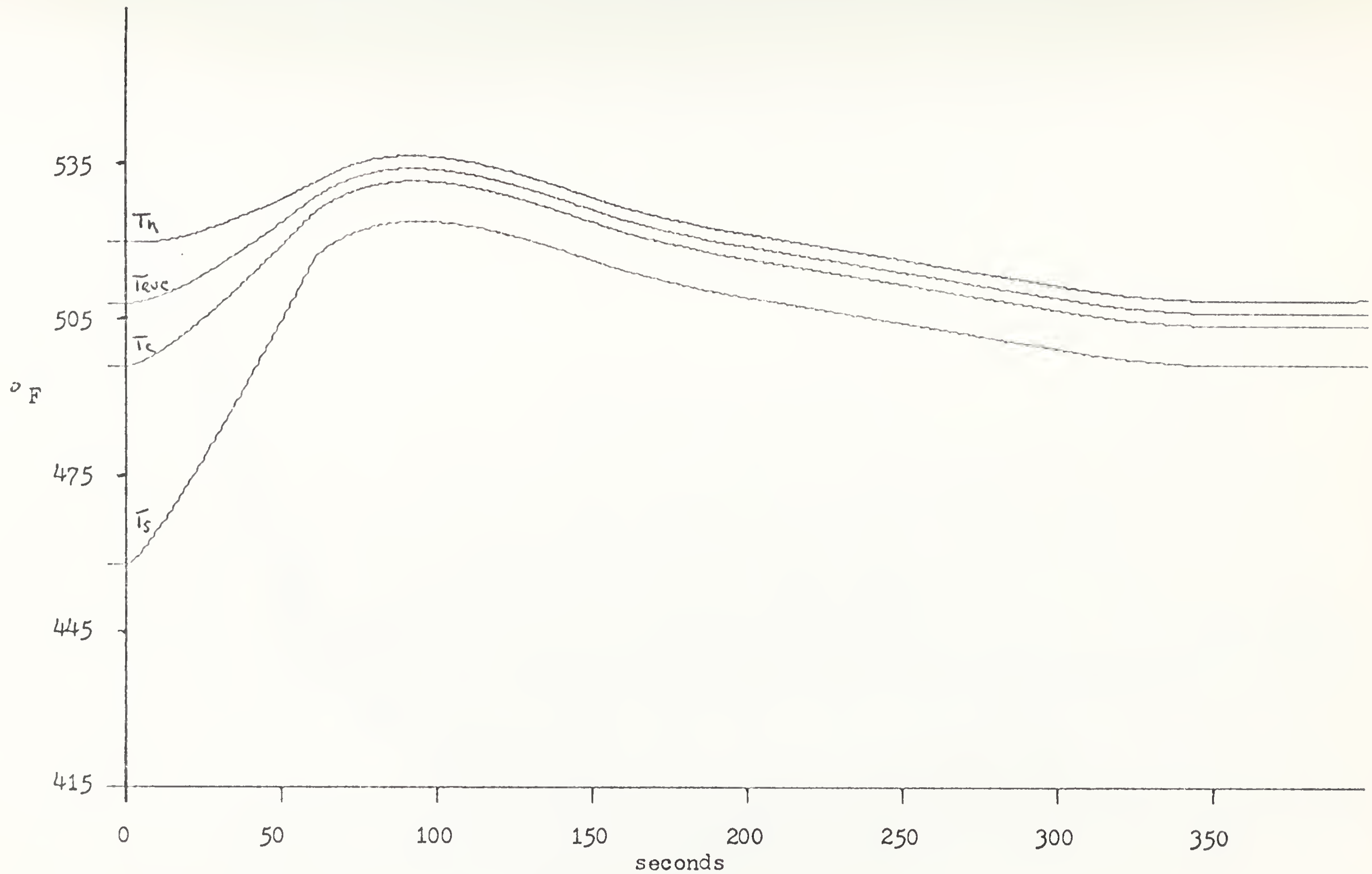

Figure 6-6 SYSTEM TEMPERATURES VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, ANTICIPATORY CONTROL

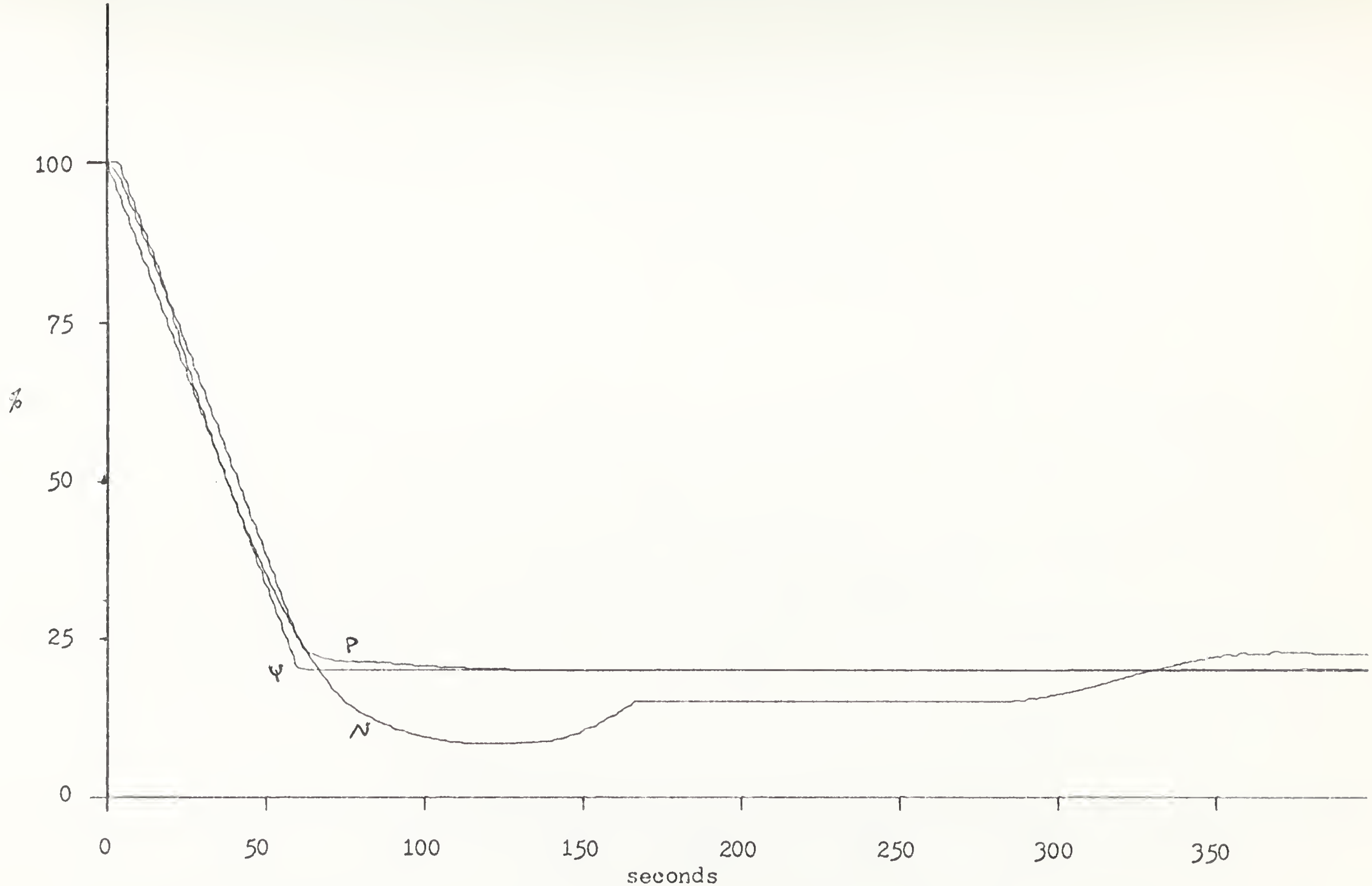

Figure 6-7 POWER LEVELS VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, ANTICIPATORY CONTROL

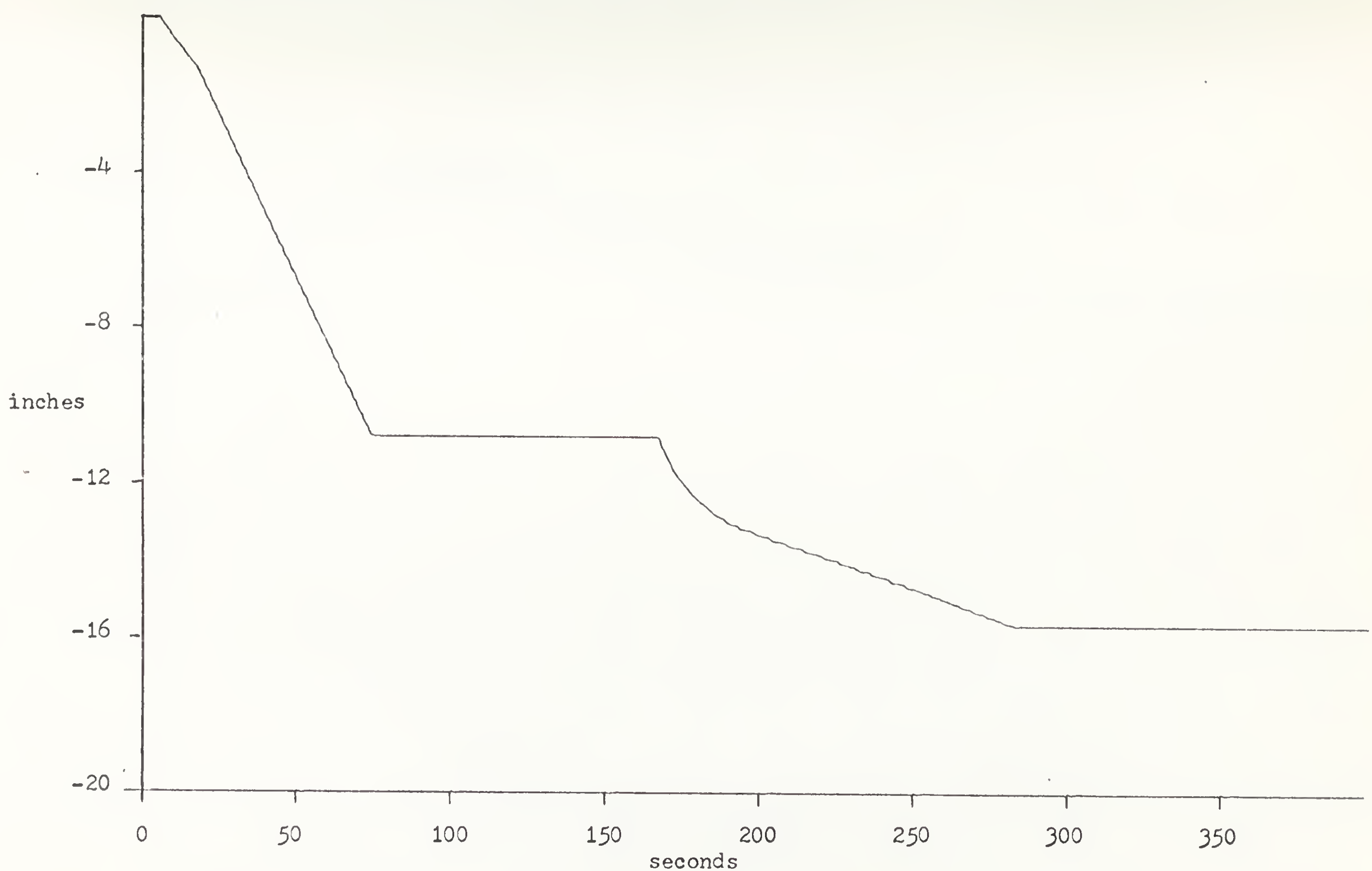

Figure 6-8 CONTROL ROD HEIGHT VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, ANTICIPATORY CONTROL

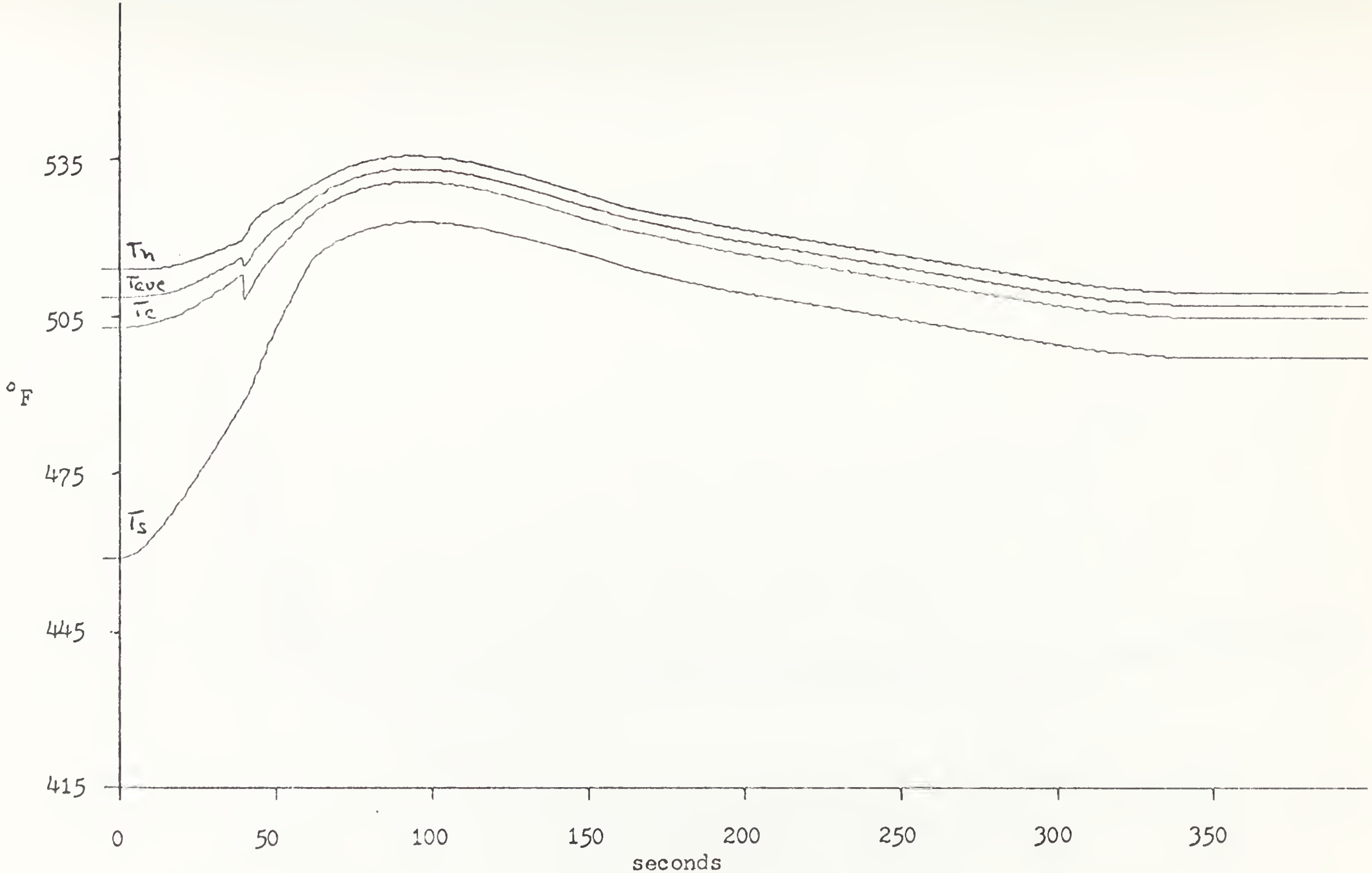

Figure 6-9 SYSTEM TEMPERATURES VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, DISCRETE FLOW CHANGE

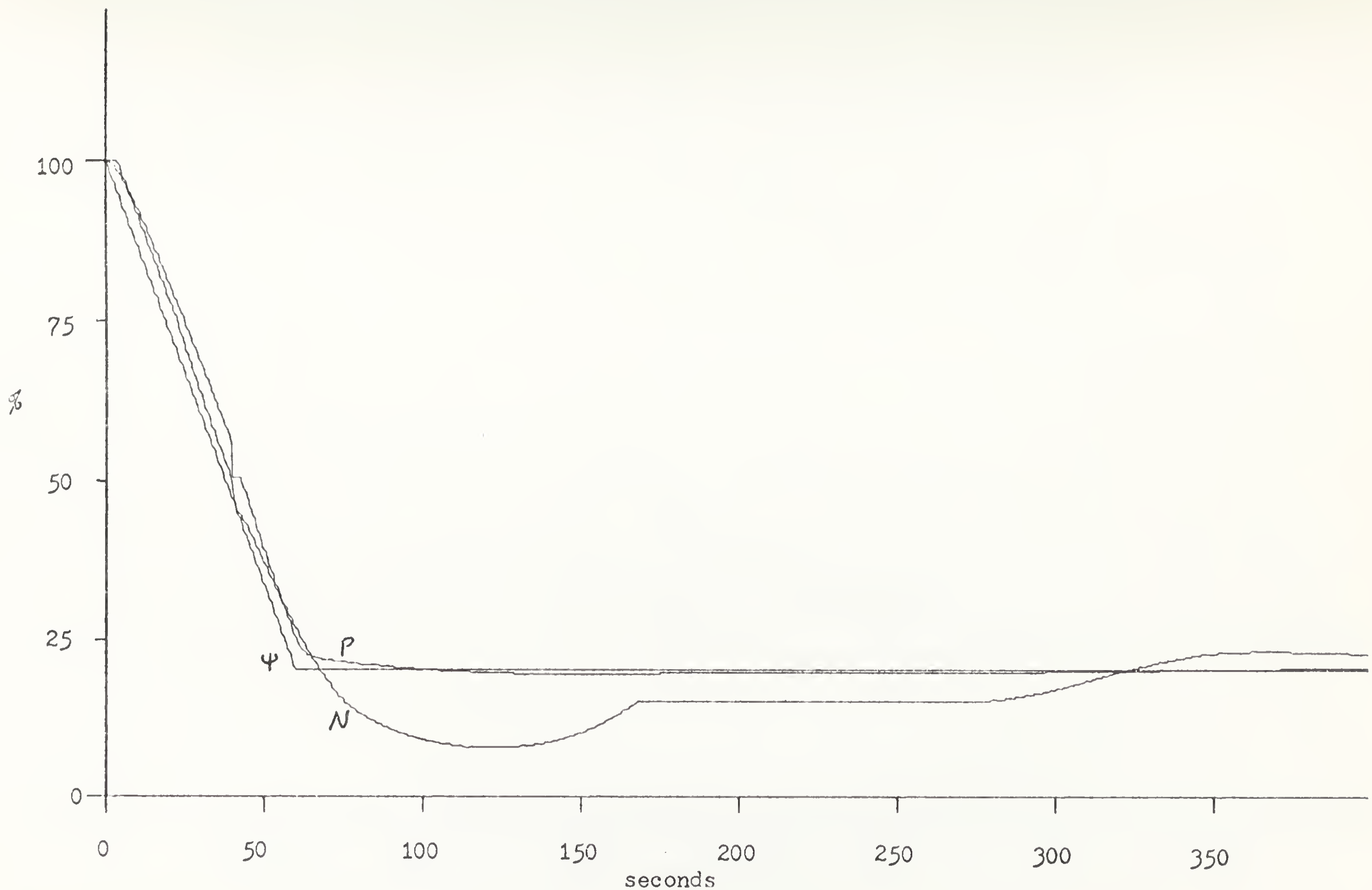

Figure 6-10 POWER LEVELS VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, DISCRETE FLOW CHANGE

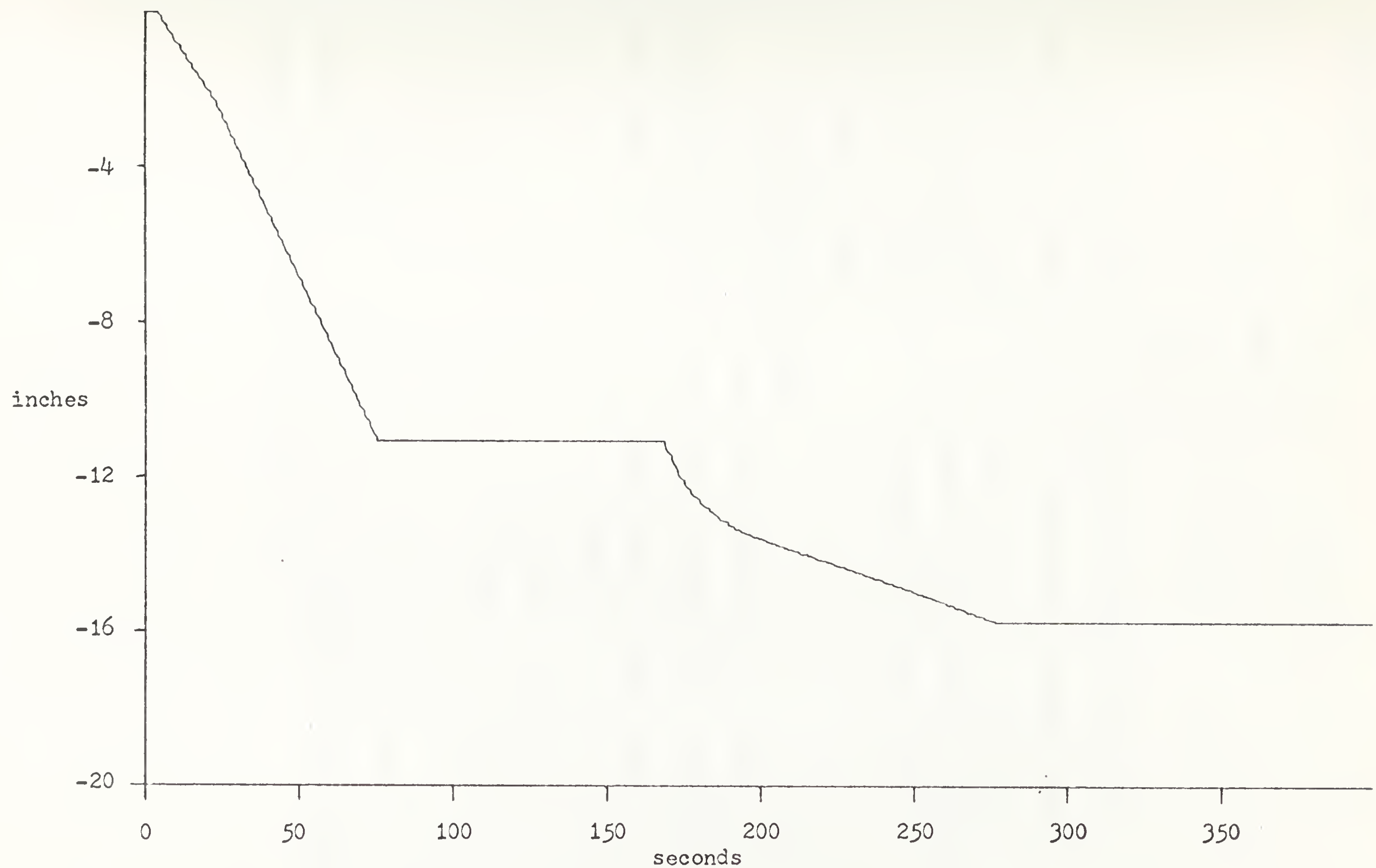

Figure 6-11 CONTROL ROD HEIGHT VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, DISCRETE FLOW CHANGE

Steady state values were:

$T_h = 510.4°F \qquad \Psi = 20.0\%$

$T_{ave} = 508.1°F \qquad N = 18.0\%$

$T_c = 505.7°F \qquad P = 19.8\%$

$T_s = 498.1°F.$

In this instance, N exceeded steam demand slightly more than in the previous case and the average initial control rod rate was 7.4 inches per minute until T_{ave} reached 511°F. Thus 15.6×10^{-4} ρ was inserted before the ten inches per minute rate was established. The second rod motion when in the linear area was at a rate of 1.59 inches per minute.

4. Continuous Flow Change

As in the up power case a flow rate of $M = 1 + \Psi$ was established and the results are shown in figures 6-12, 13 and 14. Steady state was reached in 310 seconds and the values were:

$T_h = 510.0°F \qquad \Psi = 20.0\%$

$T_{ave} = 508.0°F \qquad N = 18.0\%$

$T_c = 506.1°F \qquad P = 19.7\%$

$T_s = 498.1°F.$

T_h reached a maximum of 535.3°F and power again had a minimum of 7.7%. The control rod travel rates where insignificantly different from the discrete pump shift case.

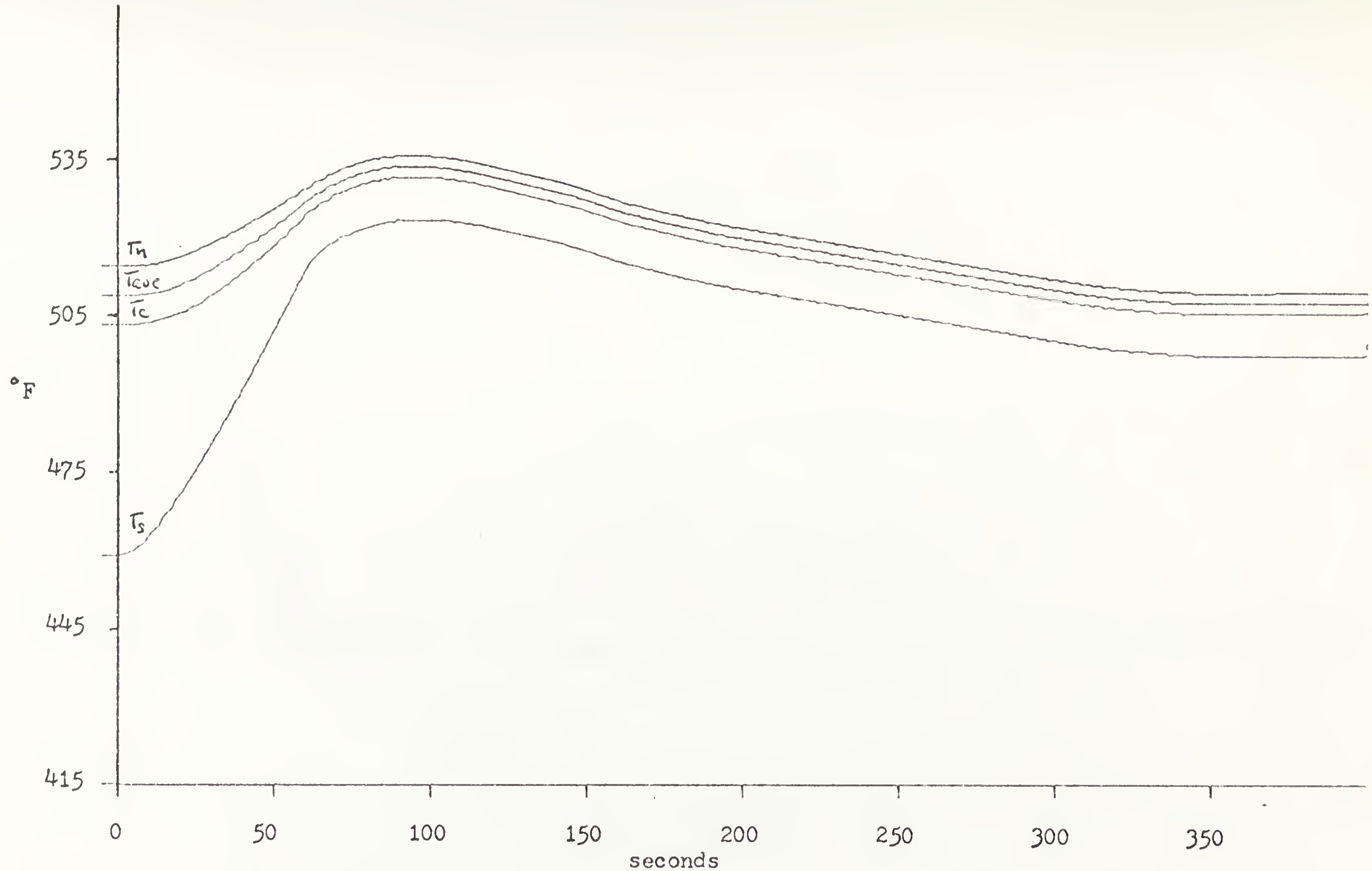

Figure 6-12 SYSTEM TEMPERATURES VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, CONTINUOUS FLOW CHANGE

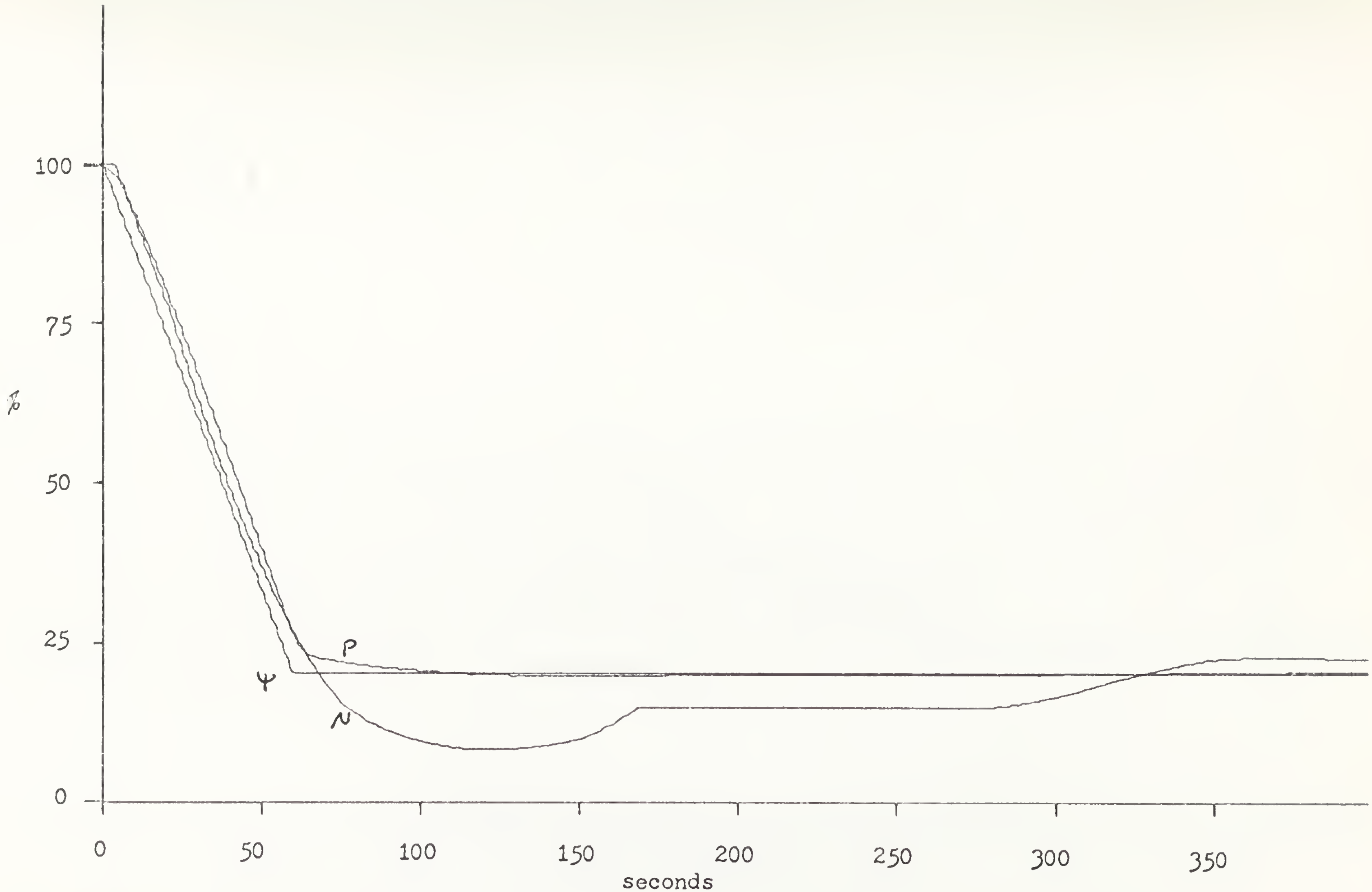

Figure 6-13 POWER LEVELS VERSUS TIME
MODIFIED DOWN POWER TRANSIENT, CONTINUOUS FLOW CHANGE

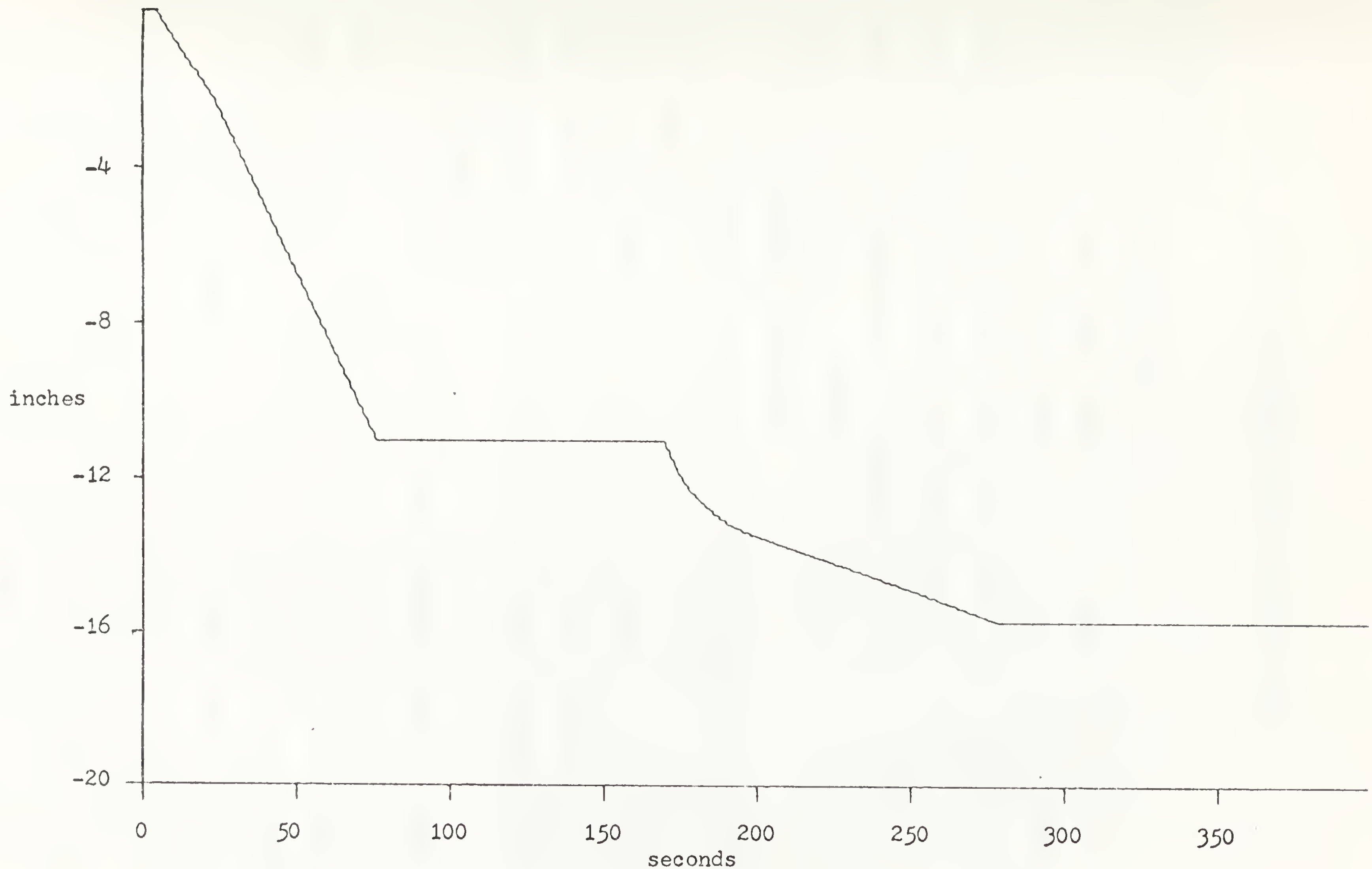

Figure 6-14 CONTROL ROD HEIGHT VERSUS TIME
MODIFIED DOWN POWER TRANSEINT, CONTINUOUS FLOW CHANGE

VII. CONCLUSIONS AND RECOMMENDATIONS

A. CONCLUSIONS

1. The Up Power Transient

In this case several specifics were noted. The sooner the initial rod withdrawal, the quicker was the response of the system. Increasing the coolant flow rate did not substantially degrade the response time but did reduce the maximum power and increased the minimum T_s. The continuous flow change system gave a slightly better response than the discrete flow change.

2. The Down Power Transient

As noted above the sooner the rod motion started the quicker the response. But in this case a discrete flow change further decreased the response time as well as the maximum T_h. The minimum power level was not affected. The continuous flow change caused a slight decrease in T_h while response time increased insignificantly.

3. Final Control

A satisfactory control for these transients was developed using only two rod speeds, twenty and ten inches per minute. A simple two speed motor for the control rod drive mechanisms could be used, greatly simplifying the variable speed used on the Savannah.

Overall comparison showed that the control using the T_{ave} band, an anticipatory rod motion response and continuous flow changes was the system which responded best to both the up power and the down power transients. But due to the complexity of variable speed motors vice dual speed the discrete flow change control would be a more practical system with only a slight reduction in response.

B. RECOMMENDATIONS FOR FURTHER STUDY

Several extensions of this investigation merit further study.

1. This study used a two speed rod control. Either a variable speed system as used on the N.S. Savannah or optimizing the rod speed to one value, thus enabling the use of a still simpler motor, could be used as a basis for further work.

2. As this study looked at the two transients separately, a study of the two maneuvers in rapid succession would provide a good measure of the real worth of the control systems.

3. An investigation could be made into the effects of allowing rod travel in excess of the steady state limit by some percentage in an effort to speed up the system response.

4. A study of the two loop response could be undertaken including the effects of coolant loop isolation or steam stop valve closure.

5. This study used equal transport delays between the reactor and the heat exchanger and vice versa. Other plants have substantially more delay in the cold leg than the hot leg and simulation of this could generate new controls.

APPENDIX

THE COMPUTER SIMULATION

Initially the reactor plant was simulated on the IBM-360 using the INTEG program.[1] The limit of 4500 points required a delta time of greater than 0.3 seconds to complete the up power transient. In fact, even using a delta time of 0.01 seconds caused over- and underflow conditions. The best delta time found was 0.005 seconds which only allowed 22.5 seconds of the simulation.

Discarding the INTEG approach, the IBM Digital Simulation Language (DSL) was selected. For all the analysis presented a delta time of 0.005 seconds was used in the fixed step size, fourth order Runge-Kutta integration. As a check a delta time of 0.001 seconds was also used which yielded less than 0.3 F difference in temperature values and less than 0.4% difference in power levels. But since the smaller delta time required over forty-five minutes of computer time vice the nine minutes for the larger delta time, the 0.005 delta time was selected to increase turn around time.

The fixed step size integration method was chosen to facilitate the calculation of control rod worth. A NOSORT/SORT block was required for the control system which precluded the use of an integration statement for CRW. Thus a fixed step size was needed to be able to generate a constant withdrawal rate. When initially running the simulation the actual rate observed was five times as

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[1]INTEG is a locally developed integration and plot package of the W. R. Church Computer Center, Naval Postgraduate School, Monterey, California.

fast as calculated. As no explanation could be found, other than a possible peculiarity of the program method, rod speeds of two and four inches per minute were put into the program resulting in ten and twenty inches per minute for the simulation.

An inconvenience with DSL was that the graphical output on the CALCOMP plotter at the Naval Postgraduate School placed the labeling for the abscissa to the right of the axis into the graph proper. This required modification of the plots for presentation as figures in this paper.

COMPUTER PROGRAM

DOWN POWER EXAMPLE

```
TITLE  DOWN POWER TRANSIENT W/DISCRETE FLOW CHANGE
CONTRL DELT=0.005,FINTIM=450
INTEG  RKSFX
INCON  TSIC=458,TCCIC=502,TCAVIC=508,TFIC=1108,THIC=514
INCON  CRW=0,CRP=0,NIC=1,N=1,CIC=1,C=1,M=2,DLTP=0.05
NOSORT
*   POWER TRANSIENT INPUT
        IF (TIME.LT.50) GO TO 1
        IF (TIME.GT.110) GO TO 2
        PSI=1-(TIME-50)*0.08/6
        GO TO 3
1       PSI=1
        GO TO 3
2       PSI=0.2
3       CONTINUE
*   FLOW CHANGE CONTROL
        IF (N.GT.0.5) GO TO 7
        M=1
7       CONTINUE
SORT
        P=0.02*(TAV-TS)
        TSDOT=70*(P-PSI)/3/M
        TS=INTGRL(TSIC,TSDOT)
        TC=((2-0.48/M)*TH+0.96*TS/M)/(2+0.48/M)
        TCCDOT=(TC-TCC)*M/6
        TCC=INTGRL(TCCIC,TCCDOT)
        TCADOT=(TF-TCAV-50*M*(TCAV-TCC))/25
        TCAV=INTGRL(TCAVIC,TCADOT)
        TFDOT=(600*N-TF+TCAV)/25
        TF=INTGRL(TFIC,TFDOT)
        THDOT=(2*TCAV-TCC-TH)*M/6
        TH=INTGRL(THIC,THDOT)
        TAV=(TH+TC)/2
        RLMT=0.0138*(PSI-1)
NOSORT
*   ROD MOTION CONTROL
        IF ((508-TAV).GT.3) GO TO 4
        IF ((508-TAV).LT.-3) GO TO 5
        IF ((N-PSI).GT.DLTP) GO TO 5
        GO TO 6
4       CRW=CRW+7E-7/3
        GO TO 6
5       IF (CRW.LT.RLMT) GO TO 6
        IF (N.LT.0.15) GO TO 6
        CRW=CRW-7E-7/6
6       CONTINUE
SORT
```

```
          KEF=1.120988-1.88E-4*TCAV-2.3E-5*TF+CRW
          NDOT=2E4*(N*(KEF-1)/KEF+75E-4*(C-N))
          N=INTGRL(NIC,NDOT)
          CDOT=0.1*(N-C)
          C=INTGRL(CIC,CDOT)
          CRP=CRW/7E-4
PRINT  2,TH,TAV,TC,TS,N,PSI,P,CRP
END
STOP
```

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